December 5, 1997

EA 97-562

Mr. Bruce D. Kenyon
President and Chief Executive Officer
Northeast Nuclear Energy Company
P.O. Box 128
Waterford, CT 06385

SUBJECT: NOTICE OF VIOLATION AND INDEPENDENT CORRECTIVE ACTION VERIFICATION PROGRAM FUNCTIONAL INSPECTION OF MILLSTONE UNIT 3 (NRC INSPECTION REPORT NO. 50-423/97-206)

Dear Mr. Kenyon:

During the periods from August 18-29 and September 8-19, 1997, the staff of the Special Projects Office (SPO) performed a safety system functional inspection (SSFI), of the emergency core cooling and seal injection functions of the chemical volume and control systems. The capability of the system to perform the safety functions required by its design basis, adherence of the system to its design and licensing bases, the consistency of the as-built configuration with the Final Safety Analysis Report (FSAR), and the consistency of system operations with the plant technical specifications were assessed. The functions of important support systems including charging pump cooling, electrical, and ventilation, and those of interfacing systems such as the recirculation spray, reactor plant component cooling, service water, and safety injection were also verified.

The SSFI was conducted as part of the NRC staff's Millstone restart review process, as described in SECY-97-003, dated January 3, 1997. Northeast Nuclear Energy Company (NNECO) has been carrying out assessments, principally through the Millstone Unit 3 Configuration Management Plan (CMP), to provide assurance that Unit 3 is in conformance with its design and licensing bases. The NRC staff's SSFI, conducted by the SPO, is part of a multifaceted effort designed to verify the effectiveness of your CMP efforts. The results of this inspection, together with additional team inspections, and results from reviews by the Independent Corrective Action Verification Program (ICAVP) contractor (Sargent and Lundy) will be used by the NRC to assess the effectiveness of your CMP.

The findings of the inspection were presented to NNECO during a public exit meeting on September 24, 1997. The inspection identified three principal findings, as discussed below and in further detail in the report. Additionally, the inspection identified other issues, which are fully discussed in the report.

The first principal finding identified was that NNECO's staff had failed to recognize and evaluate the potential for air in certain portions of the recirculation spray system to be swept into the suction of the charging and safety injection pumps during the cold-leg recirculation phase of a loss-of-coolant accident. While the results of ongoing analysis of this issue may lead to the conclusion that the air will have little or no effect on system or pump performance, the failure of NNECO's staff to fully analyze the system and understand its performance is, by itself, a concern.

The second principal finding by the inspection was the apparent inadequacy of your Technical Specification (TS)-required program to minimize leakage outside the containment. Specifically, the program addressed leakage from valve stems, piping joints, and other direct leakage pathways. However, it failed to adequately account for potential valve leakage from systems carrying highly radioactive water to the refueling water storage tank following a postulated accident. This finding also identified the apparently inadequate review (conducted by NNECO) of information on this issue contained in a 1991 NRC Information Notice (IN). The information contained in the IN, and in the existing calculations concerning post-accident leakage, provided opportunities for the NNECO staff to identify and correct the problems with the program.

The apparent violations discussed above are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," (Enforcement Policy), NUREG-1600. No response regarding these two apparent violations is required at this time; however, any corrective actions deemed appropriate should be instituted in a timely manner. Please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review or additional information received at a predecisional enforcement conference, should the decision be made to hold one. You will be advised by separate correspondence about further actions on these issues.

The third principal issue identified during the inspection was the inadequacy of the TS which requires valve lineup every 31 days for charging flow path valves not locked or otherwise secured in position. While the TS do not require that the lineup be verified using a single valve lineup procedure, the procedure reviewed during the inspection was the only one designated as necessary to comply with the TS. That procedure did not identify a complete listing of valves required to ensure proper lineup. While your staff was able to demonstrate that all the valves missing from that lineup were accounted for, either in other lineups or by position alarms, this issue raises a question about the rigor and depth of NNECO's configuration management program.

The above issues, as well as others described in the report that include inadequate procedures and failure to correct FSAR discrepancies, are indicative of potential weaknesses in your CMP and have been cited as violations in the enclosed Notice of Violation. Please note that you are required to respond to the Notice of Violation and should follow the instructions specified in it when preparing your response. The NRC will use your response, in part, to determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Based on our findings, your staff initiated an evaluation of the effectiveness of the CMP. In your response to the Notice of Violation please include a discussion of the scope and results of your evaluation of the CMP.

In accordance with Title 10 of the *Code of Federal Regulations,* Section 2.790(a), a copy of this letter and the enclosures will be placed in the NRC's Public Document Room.

Should you have any questions concerning the enclosed inspection report, please contact the project manager, Mr. J. Andersen at (301) 415-1437, or the inspection team leader, Mr. J. Luehman, at (301) 415-3150.

Sincerely,

Original signed by:

Eugene V. Imbro, Deputy Director
ICAVP Oversight
Special Projects Office
Office of Nuclear Reactor Regulation

Docket No.: 50-423

Enclosures:
1. Notice of Violation
2. Inspection Report 50-423/97-206

cc w/enclosures: see next page

cc w/enclosures:

M.H. Brothers, Vice President - Millstone, Unit 3
D.M. Goebel, Vice President, Nuclear Oversight
J.K. Thayer, Recovery Officer, Nuclear Engineering and Support
P.D. Hinnenkamp, Director, Unit Operations
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J.M. Block, Esquire, CAN
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Representative T. Concannon
E. Woollacott, Co-Chairman, NEAC

Distribution for Memorandum to B.D. Kenyon dated:

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NOTICE OF VIOLATION

Northeast Nuclear Energy Company
Millstone Nuclear Power Station
Unit 3

Docket No. 50-423
License No. NFP-49

During an NRC inspection conducted August 18-29 and September 8-19, 1997, violations of NRC requirements were identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violations are listed below:

A. Technical Specification (TS) 6.8.1.a requires, in part, that written procedures be established and maintained in accordance with the applicable portions of Appendix A of Regulatory Guide 1.33, Revision 2, February 1978. The applicable portions of Appendix A of Regulatory Guide 1.33, Revision 2 include procedures for equipment surveillance and annunciator response.

Contrary to the above, during an NRC inspection that ended September 19, 1997, the following instances were identified where the requirements of TS 6.8.1a were not met:

1. SP 3608.4, the procedure specified to verify that each charging injection flow path valve not locked or otherwise secured was in its correct position every 31 days as required by TS 4.5.2 b.2 was inadequate. Specifically, the procedure failed to require the verification of the position of 13 unlocked/unsecured charging injection flow path valves (including 3CHS*V706, 3CHS*V44 and CHS*V707).

2. Procedure SO 3604 A.5, Rev. 10, which specifies the testing required of alternate charging path valves CHS*V190 A and B was inadequate. Specifically, the procedure failed to include a prerequisite that the valves, which are only used during special circumstances such as following a fire, only be tested during cold shutdown when normal charging is isolated.

3. Procedure SP 3604C.1 which is the procedure that demonstrates that refueling water storage tank (RWST) level is compliance with TS 3.5.4a was inadequate. Specifically, the procedure as written relies only on wide-range RWST level to verify compliance with the TS but given the inadequate accuracy of the wide-range RWST level, the annunciators actuated by the narrow range level are also necessary and are used by the operators used to demonstrate compliance with the TS.

4. OP 3353.MB1C, "Main Board 1C Annunciator Response," Rev. 1, Change 5, 1-1B "CTMT Recirc CLR SW FLOW HI/LO" is inadequate. Specifically, step 6.2.2 incorrectly required the safety injection pump be stopped for a service water (SW) rupture even with the safety injection actuated.

5. OP 3353.MB1C, "Main Board 1C Annunciator Response," Rev. 1, Change 5 ,1-1A "RPCCW HX SW Flow HI\LO" is inadequate. Specifically, the procedure failed to require that after

an alarm at 6200 gpm is received that a verification of adequate SW flow (7125 gpm) to the heat exchanger for achieving safety grade cold shutdown is performed.

This is a Severity Level IV violation (Supplement I).

B. TS 4.5.2.b.1 requires that in Modes 1-4, ECCS piping, with the exception of the RSS pump, heat exchanger and associated piping be verified to be full of water by venting the ECCS pump casing and accessible discharge high points.

Contrary to the above, accessible ECCS vent valve 3SIL*V992 was not vented every 31 days in Modes 1-4 to verify the associated piping was full of water.

This is a Severity Level IV violation (Supplement I).

C. 10 CFR 50.71 (e) requires, in part, that the licensee update the final safety analysis report (FSAR) to assure that the FSAR contains the latest information developed at an interval not to exceed 24 months.

Contrary to the above, during an NRC inspection ending September 19, 1997, the following instances were identified that the information contained in the FSAR had not been updated within the past 24 months:

1. FSAR Section 6.3.3.2 states, "A makeup flow rate from one charging pump is adequate to sustain pressurizer pressure at 2250 psig [2,235 psia] for a break through a .375-inch diameter hole." However, no testing has been done to confirm that the statement and actual test data indicates that the charging pumps can not perform as described in the FSAR statement.

2. FSAR Section 9.4.3.1, Item 12, states that air flow in the auxiliary building shall be maintained from least contaminated to more contaminated spaces. However, in the winter alignment, air is recirculated from the potentially more contaminated charging pump rooms back to the less contaminated areas of the auxiliary building.

This is a Severity Level IV violation (Supplement I).

D. 10 CFR Part 50, Appendix B, Criterion XVI, requires that conditions adverse to quality be identified and promptly corrected.

Contrary to the above:

1. The licensee's corrective actions for restricting the use of Teflon tape on components that may be exposed to high radiation doses was inadequate. Specifically, in CR M3-96-0067 the licensee identified and prohibited the use of Teflon tape on components in containment as a condition adverse to quality because of the potential for high radiation doses but failed to

similarly restrict the use of such tape in areas outside the containment that could be subjected to high radiation levels.

2. FSAR Table 6.3.3, "MOV in the ECCS," which was previously identified by the licensee as requiring updating (FSARCR 97-MP3-323), was found to require further updating as the information on the interlocks associated with valve 3 CHS*8840 B was still incorrect.

3. FSAR Section 7.3.1.1.5 descriptions of the bypassed and inoperable status indication equipment, which was previously identified by the licensee as requiring updating (FSARCR 97-MP3-101), was found to require further updating as the information failed to reflect that an inoperable safety injection cooling pump would cause the safety injection pump to be inoperable.

This is a Severity Level IV violation (Supplement I).

E. 10 CFR 50.55a requires, in part, that plant systems be constructed and modified in accordance with the applicable ASME Code. The applicable version of the ASME Code, 1971 through the Summer 1973, addenda requires, in part, that the stress indices of NB-3600 be used for Class 1 piping, and the requirements of NC-7512 for pressure drop considerations and those of NC-3677 for sizing of discharge piping be followed when designing system relief valves.

Contrary to the above, when installing orifices 3SIH*R038, 039, and 041 in the charging system, the calculation supporting the modification failed to consider the proper stress indices for the installation of the offices. When designing charging system relief valves pressure drop considerations for 3CHS*RV8119 and 3CHS*RV8123 failed to consider possible volume control tank back pressure and the area of the common discharge line for 3CHS*RV8119 and 8123 was considerably smaller than the required combined area of all lines discharging to it.

This is a Severity Level IV violation (Supplement I).

F. 10 CFR Part 50, Appendix B, Criterion III, requires, in part, that design control measures be provided for verifying or checking the adequacy of design.

Contrary to the above, measures for verifying the adequacy of the design of the annulus pipe rack structure were inadequate. Specifically, design verification measures failed to identify that the 1996 calculations performed to demonstrate the annulus rack was within the design stress and load limits failed to consider the masses of the supported piping in analyzing the performance of the structure.

This is a Severity Level IV violation (Supplement I).

G. TS 6.8.1a requires, in part, that written procedures be established and implemented in accordance with the applicable portions of Appendix A of Regulatory Guide (RG) 1.33, Rev. 2, February 1978. The applicable portions of RG 1.33, Rev. 2, Appendix A, include procedures for maintenance and replacement of equipment.

Contrary to the above, during an NRC inspection that ended September 19, 1997, it was identified that certain required procedures were not properly implemented. Specifically:

1. Nuclear Group Procedure (NGP) 6.12, Rev. 1, "Evaluation of Replacement Items," requires that Replacement Item Evaluations (RIE) include preventative maintenance management system (PMMS) and bill of materials (BOM) updates. However, RIE No. PSE-MP3E-94-101, which evaluated the replacement of relief valve 3CHS*RV8119, did not include PMMS and BOM updates to reflect a change of valve sealing gasket material from asbestos
(40-year life) or Buna-N (13-year life).

2. Specification SP-ME-570, Rev. 3, "Field Fabrication and Erection of Piping and Supports requires, in part, that all bolted or stud connections have full thread engagement with the nuts. Further, Common Maintenance procedure, C-MP-715A, Rev. O, requires, in part, that it be verified that nuts have full engagement on studs or bolts. However, it was identified that the Reactor

Coolant Pump A and D seal water injection line pump connections flanges, last made up in 1993, each had nuts that were not fully engaged to their respective studs.

3. Procedure CC1, Rev. 2, "Control of Chemical Consumable Products" requires, in part, that chemical consumable products, including tape, used on primary or secondary systems are either labeled "A" for chemical product use category or are controlled by instructions in the automated work order. However, during a walkdown of the seal water injection piping, yellow plastic high pressure tape, not controlled as required above, was found attached to the piping.

This is a Severity Level IV violation (Supplement 1).

H. 10 CFR 50.49 requires, in part, that safety and nonsafety-related equipment whose failure under postulated environmental conditions could prevent the reactor from being shut down and maintained in a safe shutdown condition shall be included in the licensee's environmental qualification program and appropriate environmentally qualified.

Contrary to the above, during an NRC inspection that ended September 1,1997, it was identified that motor operators for valves 3MM*-MOV-18A-D, equipment whose failure under postulated environmental conditions could prevent the reactor from being shut down and maintained in a safe shutdown condition, were not in the licensee's environmental qualification program or appropriately environmentally qualified.

This is a Severity Level IV violation (Supplement 1).

Pursuant to the provisions of 10 CFR 2.201, Northeast Nuclear Energy Company is hereby required to submit a written statement or explanation within 30 days of receipt of the letter transmitting this Notice of Violation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, with a copy to the Director, Special Projects Office, Office of Nuclear Reactor Regulation, and a copy to the NRC Resident Inspector at the Millstone Nuclear Power Station, Unit 3. This reply should be clearly marked as a "Reply to a Notice of Violation," and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations,
and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the required time specified in this

Notice of Violation, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

Dated at Rockville, Maryland
this 5th day of December, 1997

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Report No.:	50-423/97-206
Docket No.:	50-423
License No.:	NPF-49
Licensee:	Northeast Nuclear Energy Company
Facility:	Millstone Unit 3
Location:	Millstone Nuclear Power Station 156 Rope Ferry Road Waterford, Connecticut 06385
Dates:	August 18–29 and September 8-19, 1997
Inspectors:	James Luehman, ICAVP, Leader, Sub-Team 3A Special Projects Office Stephen Tingen, Special Projects Office Donald Prevatte, Mechanical Engineer, Contractor* Robert Quirk, I&C Engineer, Contractor* Omar Mazzoni, Electrical Engineer, Contractor* Victor Ferrarini, Mechanical Engineer, Contractor* *Contractors from Parameter, Inc.
Approved by:	Peter Koltay, ICAVP, Leader, Team 3 Special Projects Office Office of Nuclear Reactor Regulation

Enclosure 2

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EXECUTIVE SUMMARY
Millstone Nuclear Power Station
Inspection Report 50-423/97-206

During the periods from August 18-29, and September 8-19, 1997, a team from the U.S. Nuclear Regulatory Commission's (NRC) Special Projects Office, Office of Nuclear Reactor Regulation, conducted a Safety System Functional Inspection (SSFI) at Millstone Nuclear Power Station Unit 3. The inspection was done as a part of the NRC's oversight of the Independent Corrective Action Verification Program (ICAVP). The purpose of this inspection was to assess the effectiveness of the licensee's Configuration Management Plan (CMP) identifying areas of nonconformance with the plant's design and licensing bases by inspecting a system that was not in the scope of the ICAVP contractor's review. The team's review focused on the emergency core cooling and seal injection functions of the charging system.

The team identified apparent violations and other findings that indicated that the licensee's configuration management program efforts undertaken, in part, in response to the findings of the 1996 NRC "Special Inspection of Engineering and Licensing Activities at Millstone Nuclear Power Station" (Report 50-423/96-201), failed to identify some potentially significant licensing and design basis issues.

The team's principal findings were identified in the areas of operations, surveillance, and maintenance. The team found that the air in certain piping in the initially dry recirculation spray system (RSS) could be swept into the suctions of the charging and safety injection pumps during the cold-leg recirculation phase of a loss-of-coolant accident (LOCA). The potential for air binding these pumps was recognized by Northeast Nuclear Energy Company (NNECO) prior to initial plant startup. However, that review failed to completely identify and analyze the extent of the problem. Given the somewhat unique configuration of the RSS (an initially dry system subsequently providing net positive suction head for two water-filled systems) the team was concerned not only that the licensee had initially failed to fully identify this issue but, even after the issuance of NRC Information Notice (IN) 88-23, "Potential for Gas Binding of High Pressure Safety Injection Pumps," and its four supplements, this issue still went unidentified.

The licensee's program for minimizing leakage outside the containment (a requirement of Technical Specification (TS) 6.8.4) did not address inter-system leakage that could result in radioactive water leaking into places where radioactivity could be vented to the atmosphere, such as the refueling water storage tank (RWST) following a LOCA. A minor leakage through such valves could result in unacceptable control room and offsite radiation doses in the event of an accident. The failure to test the valves in question also is indicative of a weakness in the licensee's American Society of Mechanical Engineers (ASME) Code Section XI valve testing program.

The valve lineup procedure for verifying the TS requirement that all charging pump injection flow path valves, not locked or otherwise secured in position, are in their correct positions every 31 days was found to be inadequate. For this specific case, the licensee was able to subsequently show, by use of other valve lineup procedures and reliance on valve position alarms, that it was unlikely that any of the valves not included in the designated lineup procedure could in fact remain out of position.

A concern in the structural design area was the licensee's failure to account for the post-accident temperature rise inside containment on rigidly restrained structural steel. In an elevated temperature environment the steel will tend to expand more rapidly than the concrete around it. Excessive expansion of the steel could result in damage to the concrete structures as well as consequential damage to nearby equipment.   At the end of the inspection the licensee was evaluating the potential effects of this issue.

In the area of mechanical design, the removal of certain main steam valves from the electrical equipment qualification program was identified as an issue. Originally, the main steam block valves in question were part of the environmental qualification program. Subsequently, questions relative to the actual qualification of the valves arose and the licensee performed an analysis that justified the removal of the valves from the program. The team concluded that based on the available documentation, the removal of the valves from the program was not properly justified.

The lack of adequate justification for the qualification of the seal material used on certain RSS containment isolation valves was an unresolved item. At the end of the inspection the licensee had not provided the team with sufficient information to support its position that the material was qualified for the radiation exposure it would receive during a design basis LOCA.

In the areas of instrumentation and control and electrical design the team identified weaknesses in calculation control, control of documentation of assumptions supporting calculations, and inadequacies in the consideration of instrument uncertainties.

Overall, the team found the material condition of the charging system and associated support systems to be good. The significant engineering calculation work ongoing, the outstanding testing of a major orifice modification to the charging system, and the outstanding issues the team identified on the RSS (which supports the charging system in the recirculation phase of an accident), precluded the team from verifying that the system would perform its intended function under all design conditions. However, with all modifications completed, and test results found to be acceptable, the system will perform its intended function.

The team identified more findings than would be expected following an effort such as the licensee's CMP. Specifically, issues such as the effects of air entrainment in RSS piping, inadequate inter-system leakage monitoring of valves, and an example of a valve lineup procedure that failed to meet TS requirements, were expected to be identified through the licensee's CMP efforts. Additionally, at least some of the less significant procedural and FSAR inconsistencies noted by the inspection team, should also have been identified and corrected by the licensee's CMP.

The strengths identified during the inspection included an aggressive maintenance program, knowledgeable system engineers, and a document control organization that appears to have a good understanding of the programmatic challenges that remain in improving the unit's calculation control program.

1.0 Introduction

On August 18-29 and September 8-19, a team from the U.S. Nuclear Regulatory Commission's (NRC) Special Projects Office, Office of Nuclear Reactor Regulation, conducted a Safety System Function Inspection (SSFI) at Millstone Nuclear Power Station. The inspection was done as part of the NRC's oversight of the Independent Corrective Action Verification Program (ICAVP) at Millstone. The purpose of this inspection was to provide the NRC with additional insights to evaluate the effectiveness of the licensee's Configuration Management Plan (CMP).
The system selected for the inspection was outside the scope of the ICAVP contractor's review but was a system for which CMP was completed.

1.1 Background

The safety and seal injection functions of the charging system were selected for this inspection from among the 88 Groups 1 and 2 systems covered by Title 10 of the *Code of Federal Regulations,* (10 CFR) Section 50.65, also known as the Maintenance Rule. All 88 Groups 1 and 2 systems were evaluated by the licensee's CMP. The specific criteria that were considered in choosing the charging system included the relatively high number of other risk significant systems it interfaces with, the number of modifications and changes made to the system since plant licensing, and the relatively high risk significance the system has when considering accident scenarios.

1.2 System Description and Safety Function

During normal operations, the charging system has numerous functions including providing makeup water to the reactor, seal injection cooling for the reactor coolant pumps (RCP), and boric acid for reactivity control. During and after an accident, the charging system provides RCP seal cooling, the high-pressure injection of borated water from the RWST during the injection phase, and one of the pathways for recirculation of containment sump water during later stages of an accident. It is this accident-mitigation group of functions that was the focus of the inspection.

1.3 Inspection Scope and Methodology

The team used SSFI, Inspection Module 93801, as its inspection approach. Because the purpose of the inspection was to assess the effectiveness of the licensee's CMP, the emphasis of the inspection was system design, including modifications, and the appropriate translation of the design and licensing bases into operations and surveillance activities.

1.3.1 Millstone Unit 3 Charging System Inspection Boundaries

Since the mechanical, electrical, control, and structural aspects of charging system interface with other plant systems, the inspection included some portions of interfacing systems. The team identified the charging system inspection boundaries as follows:

- Seal Injection - From the charging system to containment isolation valve CHS*MV8100 and relief valve CHS*RV-8121. (RCP seal 2 and 3 leakoff piping and the No. 3 seal supply piping, and relief valve discharge piping to the pressurized relief tank (PRT) were not within the scope of this inspection).

- Emergency core cooling system (ECCS) Injection and Recirculation - The three charging pumps; piping and valves bounded by the volume control tank (VCT) outlet (MOV CHS*LCV-112B): charging isolation valve CHS*MV-8105; charging pump mini-flow line relief valves CHS*RS-8510 A and B; charging pump suction recirculation supply valves SIL*MV-8804A and B; charging system cold-leg injection connections to reactor coolant system (RCS) loops 1-4; charging pump miniflow isolation valve CHS*MV-8110; boric acid connections to the charging pump suction piping; the seal water heat exchanger connection to the charging pump suction piping; the charging system header drain and test lines bounded by SIH*CV-8842 and 8843; and the hydrostatic test pump discharge piping connection to the charging system piping.

- Support Systems - Charging pump cooling (CCE) (bounded by the CCE expansion tank and heat exchanger service water (SW) inlet/outlet valves SWP*V-63B, V64B, V31A and V32A); ventilation supply/exhaust for the charging pump rooms (bounded by dampers HVR*MOD-50A, B, C1, C2 and HVR*MOD49A, B, C1, and C2); ventilation to the auxiliary building, switchgear, and engineered safeguards features motor control centers (MCCs) for electrical components within the scope of the inspection; electrical components within the system boundaries back to the MCCs including emergency diesel generator (EDG) loading; and instrumentation within the system boundaries back to the initiating device or component.

2.0 Mechanical

2.1 Charging System

2.1.1 Scope of Review

The mechanical engineering review was aimed primarily at verifying that the charging system was capable of accomplishing its design functions, meeting all applicable codes, standards, regulatory requirements, and good engineering practices, meeting all licensing basis commitments; operating consistent with its design basis, and being tested in a manner to accurately reflect its condition and its ability to perform its intended function.

The interfacing and supporting systems for the charging system were reviewed to ensure that they could provide the necessary support for the charging system to perform its designed functions. The interfacing and supporting systems included the service water (SW), sealing and lubrication, instrument air, ventilation, reactor coolant, and recirculation spray.

During the mechanical and engineering review, special attention was focused on verifying the system and supporting systems could meet the minimum performance requirements; reviewing the systems' single failure design integrity; verifying fulfillment of the Final Safety Analysis Report (FSAR) and TS commitments; and verifying adequate piping design temperatures and pressures.

2.1.2 Findings

(a) Inadequate charging pump performance testing acceptance criteria to verify meeting FSAR small leak safe shutdown statement

FSAR Section 6.3.3.2, "Loss of Reactor Coolant from Small Ruptured Pipes, from Cracks in Large Pipes, or from the Ejection of a Control Rod which Actuate the Emergency Core Cooling System," provides the following discussion of the charging system's safety function and capability for small reactor coolant system (RCS) breaks, "Ruptures of small cross-sections cause expulsion of the coolant at a rate which can be accommodated by the charging pumps which would maintain an operational water level in the pressurizer, permitting the operator to execute an orderly shutdown. A makeup flow rate from one charging pump is adequate to sustain pressurizer level at 2,250 psia [2,235 psig] for a break through a 0.375-inch diameter hole. This break results in a loss of approximately 17.5-lb/sec [127 gpm]."

According to TS 3.4.6.2.e, "Reactor Coolant System Operational Leakage," the total flow a charging pump would have to deliver would be 127 gallons per minute (gpm) for the leak, plus 40 gpm to the RCP seals, plus at least 60 gpm through the minimum flow bypass line (the required minimum flow per SP 3604A (B & C).1, Rev. 9, dated July 28, 1997, "Charging Pump A (B & C) Operational Readiness Test") for a total of 227 gpm. The required developed head, neglecting elevation differences, which would be relatively minor, would be the RCS pressure of 2,235 psig plus the charging system resistance.

TS 4.5.2.f.1, "Emergency Core Cooling Systems Surveillance Requirements," establishes the charging pump surveillance testing performance limit of ≥ 2,411 psid developed head at recirculation flow conditions, and surveillance procedure SP 3604A (B & C) 1, Rev. 9, dated July 28, 1997, "Charging Pump A (B & C) Operational Readiness Test," was intended to test this requirement. Both of these were intended to assure that the pumps' minimum performance was above the "Accident Analysis Minimum Acceptable Curves" contained in procedure EM 31121, Rev. 6, dated December 10, 1996, "IST Pump Operational Readiness Evaluation," which were derived from the Westinghouse accident analyses.

The team's review of these documents revealed that, even when the charging system resistance was neglected, the point representing the above quoted FSAR statement (i.e, 227 gpm at 2,235 psid, when plotted on the charging pump minimum performance curve, was approximately 217 psig above the curve, and approximately 173 psig above the curve that could be projected from the TS minimum allowable point. Although a calculation that provided the charging system resistance along the normal flow path was not located, the accident injection flow path would be similar. The calculation for that flow path indicated a system resistance of about 104 psi under these flow conditions. The acceptance criteria specified in TS 4.5.2.f.1 were found, however, to be adequate to demonstrate the charging pumps' ability to meet the accident analyses performance requirements, which were considerably lower than the FSAR statement.

The team reviewed the latest actual pump performance test data. The "A" pump had the best performance, and the actual data from a May 19, 1997, test, indicated that with virtually any system resistance taken into account, and without accounting for instrument uncertainty, it would not have met the FSAR statement of 127 gpm at 2250 psia.

The pump performance specified by the FSAR statement does not appear to be required to comply with any regulatory requirement and, as discussed above, pump performance appeared adequate to meet accident analysis and ASME Code testing requirements. Therefore, the safety significance of this finding is low. Nevertheless, the plant (as-built) does not meet the FSAR.

The deficiency above represents one instance in which the Millstone Unit 3 FSAR was not maintained up-to-date or did not reflect the actual plant configuration or operating practice. As such, it constitutes a violation of 10 CFR 50.71(e). (VIO 423/97-206-01)

(b) Incomplete relief valve replacement item evaluation

Relief valve 3CHS*RV8119 was replaced with a different valve by Replacement Item Evaluation (RIE) No. PSE-MP3E-94-101. This relief valve provided overpressure protection for the letdown piping downstream of the pressure control valve that is downstream of the letdown heat exchanger. It was replaced because the original valve leaked, and it was no longer available from the vendor in its original design configuration.

One of the design features of the replacement valve that is different from the original was the sealing gasket material. The original sealing gasket material was asbestos and the replacement was Buna-N rubber. The original asbestos material was qualified for the life of the plant; however, the Buna-N material was qualified for 13 years at expected operating conditions.

Nuclear Group Procedure (NGP) 6.12, Rev. 1, "Evaluation of Replacement Items," is the licensee's procedure that governs such replacements, and it required that RIEs includes a Preventative Maintenance Management System (PMMS) update to identify material with finite lives that would require periodic replacement, (i.e., the Buna-N gasket in this valve). It also requires a bill of materials (BOM) update. The RIE contained neither update and consequently neither change required was performed by the licensee.

In response to this finding, the licensee initiated Condition Report (CR) M3-97-3105. Because this valve provides overpressure protection for the piping and has no direct accident mitigation function, its failure potential as a result of seal deterioration has little safety significance.

The failure to update the PMMS and the BOM, as required, is considered one example of a violation of Technical Specification 6.8.1, failure to follow procedures. (VIO 423/97-206-02)

(c) System relief valves' discharge path does not meet the ASME code

ASME Boiler and Pressure Vessel Code, 1971, Section III, "Nuclear Power Plant Components," Article NC-7000 requires that "Vessels, piping, valves, and pumps which are pressure-containing components of the systems within the scope of Subsection NC shall be protected from pressure and (coincident) temperature that are in excess of the design conditions."

Relief valve 3CHS*RV8119 was intended to provide this protection for the letdown piping and components downstream of pressure control valve 3CHS*PCV131. This valve's pressure was set at 300 psig per the design pressure requirements this piping. Relief valve 3CHS*RV8123 was intended to protect the RCP seal leakoff piping and components upstream of the seal water heat exchanger. Its set pressure of 150 psig is also per the piping's design pressure requirement. The discharge piping for both of these relief valves combines into a common line connected to the VCT. This line and the VCT are also protected by relief valve 3CHS*RV8120 set at 85 psig. (The VCT normally operates at approximately 15 psig.) 3CHS*RV8119 has a 4- inch discharge, and 3CHS*RV8123 has a 3-inch discharge; the common discharge line into the VCT necked down from 4 inches to 3 inches.

Article NC-7512 of the ASME Code, "Pressure Drop Considerations," states, "In determining the set pressures and capacities required to comply with these rules, full account shall be taken of the pressure drop on both inlet and discharge sides of the pressure-relief device at full flow condition. In addition, back pressure arising from discharge to closed storage or dissipation systems or from discharge of other pressure-relieving devices through common discharge piping, shall be taken into account." Article NC-3677.3, "Discharge Piping from Pressure-Relieving Safety Devices," subparagraph (d), states, "It is recommended that individual [relief valve] discharge lines be used but if two or more reliefs are combined, the discharge piping shall be designed with sufficient flow area so as not to affect operation of the relief devices and in no case shall the area of such common lines be less than the combined area of all lines discharging into it."

Contrary to Article NC-7512, the back pressure that relief valves 3CHS*RV8119 and 3CHS*RV8123 would discharge to the VCT pressure could be as high as 85 psig. Therefore, when considering a 10 percent accumulation, the relieving pressures in the lines being protected could be as high as 415 psig and 250 psig, well above the ASME Code allowable for maximum relieving pressures of 330 psig and 165 psig, respectively. Contrary to Article NC-3677.3, the area of the common 3-inch discharge line for these two valves is significantly less than (approximately 1/3 of) the combined area of the two discharge lines.

In response to the common discharge line flow area concerns the licensee stated that the ASME Code requirement related to multiple relief valves that protect the same component for a common event, and that these valves were designed to lift because of unrelated events. Therefore, it is not assumed that the relief valves lift concurrently. The licensee had no analysis to support its position that no single event could challenge both valves simultaneously. Because the common discharge piping necked down to 3 inches, thereby not even providing full flow area for relief valve 3CHS*RV8119 which has a 4-inch discharge area, the licensee's position does not adequately address the team's piping area concern and fails to address the back pressure concern.

Compliance with the ASME Code aside, the safety significance of this finding appears to be minimal. Although the design may have allowed the piping to be overpressurized, it is not likely that a failure would occur.

Failure to meet the code required size for the charging system relief valve discharge piping is an apparent violation of 10 CFR 50.55a. (VIO 423/97-206-03)

(d) Non-control of prohibited materials in plant

While inspection the charging system seal injection piping to RCP "A," the team observed a yellow plastic high-pressure (HP) tape applied to the piping. Chloride and fluoride containing materials, such as this tape, that comes in contact with stainless steel piping, in combination with elevated temperatures and stresses, can cause cracking and stress corrosion of the piping. Procedure CC1, Rev. 2, dated April 14, 1997, "Control of Chemical Consumable Products," recognizes this fact. In a note preceding Step 1.6.2, "The restriction on chemical consumable product use on corrosion-resistant metal surfaces of reactor plant systems in this procedure are to prevent induction of corrosion of nickel alloys by ingredients or contaminants of chemical products used on the metal surface when the metal surface is exposed to high system temperature and pressure," and in Step 1.6.2, "IF job involves application of chemical consumable product to a corrosion-resistant, pressure retaining metal surface of a primary or secondary system, PERFORM the following: ENSURE product is labeled "A" for Chemical Product Use Category OR ENSURE compliance with restrictions Chemical Consumable Restricted Use Permit in AWO." Attachment 2, "Controlled Chemical Consumable Products," listed "nuclear grade tapes" as a type of chemical consumable product to which the administrative controls of the procedure applies. Additional direction was provided by a memorandum from the Chemical Control Coordinator, to "Distribution," dated July 13, 1994, states that "these tapes [yellow HP tape and silver duct tape] are not to be used on primary/secondary systems."

Subsequently, the licensee issued condition report ME-97-2748 to document and evaluate this condition, and Trouble Report (TR) No. 22M3075644, to clean and inspect the affected area. The tape appears to be newly installed (i.e., since the last plant shutdown, and, therefore, would not have been exposed to high temperatures, and the piping would not have been exposed to high stresses). Therefore, all the necessary conditions to initiate cracking or stress corrosion did not yet exist.

The use of tape on primary systems is considered an example of a violation of Technical Specification 6.8.1, failure to follow procedures. (VIO 423/97-206-02)

(e) Incomplete nut-to-stud thread engagement

Specification SP-ME-570, Rev. 3, dated December 21, 1995, "Field Fabrication and Erection of Piping and Supports," page 1B-3, requires, "All bolted or stud connections shall have full thread engagement with the nuts." Common Maintenance Procedure C-MP 715A, Rev. 0, dated February 25, 1994, "General Practices for Threaded Fasteners," requires in Section 4.13, "Final

Inspection and Documentation [of bolted joints]," VERIFY nuts have full engagement on studs or bolts.

During the walkdown of the charging system, the team discovered two instances where flanged piping joints did not have full thread, stud-to-nut engagement. The first was on RCP "A" at the seal water injection line pump connection flange where two of the four studs were not fully engaged. The second was on the seal injection line connection for RCP "D" where one of the four studs was not fully engaged. The last time these joints had been assembled was when the RCPs were replaced in 1993.

Subsequently, the licensee inspected these joints and found that in every case full engagement was lacking by less than one thread, and in both cases, the nuts were of heavy grade. Their engineering evaluation determined that since the load on bolted joints is carried by the first few threads, the structural integrity of the joints was not degraded. Therefore, the team agreed that there was little safety significance of this finding.

The lack of adequate stud-to-nut engagement is another example of a violation of Technical Specification 6.8.1, failure to follow procedures. (VIO 423/97-206-02)

(f) Single failure of charging system check valves

The team identified two instances where a single component check valve of the ECCS flow paths of the charging system was not completely redundant. One of these valves was check valve 3CHS-V261 in the single common suction line from the RWST and the other check valve 3CHS-V5 in the single common injection line. During the injection phase of a LOCA, the active failure of either of these valves to open would incapacitate the system, and during the recirculation phase, the passive failure of 3CHS-V5 could also totally block system flow (V261 would not be in the flow path during the recirculation phase). This design appears to be contrary to that discussed in the FSAR.

Section 3.1 of the FSAR, "Conformance With NRC General Design Criteria," states that "the Applicants conclude that MP3 fully satisfies and complies with the [single failure criteria] GDC." Section 3.1.1.2, "Definition of Terms Used in Single Failure Criteria," defined an active failure as "one in which mechanical movement must occur to complete the component's intended function. An active component failure is failure of the component to complete its intended function upon demand." One of the examples of an active failure provided in FSAR was "the failure of a check valve to move to its correct position." "The ECCS is designed to accept a single active failure following the incident without loss of its protective function." Under the heading of "Passive Failure Criteria" it states, "Adequate redundancy of check valves is provided to tolerate failure of a check valve during the long-term as a passive component." This section also states, "Two trains of pumps, heat exchangers, and flow paths are provided for redundancy." Although the licensee initiated CR M3-97-2140 on July 11, 1997, to address single failure issues, that CR originally addressed only passive failures but was later expanded to generally address active failures as well.

In response to this finding, the licensee maintained that only power-operated valves were required to be considered for active failure. The licensee implied that the check valves at issue were "especially qualified for service" and, therefore, exempt from active component failure per FSAR Section 3.1.1.3, "Application of Single Active Failure Criterion." An "especially qualified" valve was a valve with some special feature or operating condition that would effectively preclude active failure. This section cited the safety injection accumulator check valves as examples of "especially qualified" check valves, and it stated, the post-LOCA high, differential, pressure that would ensure their opening. However, this reason was not applicable to the charging check valves since they would not necessarily experience high, differential, pressure for all accident conditions, particularly, for small-break LOCA where the differential pressure would be relatively low. No other special feature or condition was identified. Finally, the licensee maintained that it was Westinghouse's position that single failure of check valves did not have to be considered and that the FSAR statements were unintentional and incorrect.

Correcting the inconsistency between the plant as-built and the initial design by deleting the FSAR statements, as suggested by the licensee’s CR needs to be evaluated. This is identified as an Unresolved Item. (URI 423/97-206-04)

(g) Other concerns

(1) The following are examples of minor problems with calculations and procedures where the overall reliability of the documents was not affected. In some instances, multiple calculations existed that addressed the same subject with no apparent hierarchy or indication of obsolete or invalid calculations being superseded.

Calculation P(R)-0983, "NPSH Evaluation for ECCS Pumps RHS, SIH, CHS - Maximum Safeguards," Rev. 0, dated April 23, 1984.

Calculation 294, "NPSH Available for ECCS Pumps," Rev. 4, dated September 9, 1985.

Calculation 3-ENG-181, "Determination of Available NPSH to Charging Pumps During Gravity Boration," Rev. 0, dated November 14, 1990.

Calculation P(R) 0982, Rev. 0, "ECCS Pressure Drop Calculations Based on Westinghouse Piping Resistance Criteria."

Calculation P(R) 0982, "ECCS Pressure Drop Calculations Based on Westinghouse Piping Resistance Criteria," Rev. O.

Calculation RFS-P-1515, "Performance of Modified 4 Loop ECCS," Rev. 1, dated January 17, 1973.

Calculation 12179-US(B)-311, "RSS Branch Flow Analysis with Degraded Pump Curve," Rev. 0, Change 1, dated April 18, 1997.

Calculation 12179-US(B)-245, "Branch Flow Rate Analysis for Safety Injection and Containment Recirculation System," Rev. 0, Change 3, April 18, 1997.

Calculation UR(B)-393-0, "Lifetime Radiation Dose to 3RSS*MOV20A, B, C, and D," Rev. 0, dated April 30, 1984.

Calculation UR(B)-400, "Gamma and Beta Equipment Qualification Doses for Normal Operations, Depressurized LOCA and Pressurized LOCA," Rev. 0.

Although the licensee had identified, on a generic basis, that there was inadequate control of multiple calculations on the same subject, all of the calculations identified above except for the first two relating to charging pump net positive suction head (NPSH) had not been previously identified as specific examples by the licensee.

(2) The team identified errors in the following calculations and procedures.

(a) Calculation SE/FSE-C-NEU-0154, Rev. 0, Change 1, dated June 3,1997, "Millstone Unit 3 ECCS Evaluation - Future Tech Spec Change Basis."

Assumption 1 of the calculation regarding hydraulic resistances was in error. If the predicted resistances were 30 percent higher than actual, reducing them back to the actual resistance values would require a global reduction factor of 76.9 percent, not 70 percent. Using a reduction factor of 70 percent would reduce the resistances to 91 percent of the actual values, which would produce predicted flow values higher than actual and be non-conservative with respect to the 10 CFR 50.46 accident analyses.

The safety significance of this finding is minimal. The flow model was used to predict flow orifice sizing and the throttle valve positions associated with a recent modification where flow orifices were installed to eliminate excessive throttle valve erosion and to prevent system runout during the recirculation phase of an accident when the charging pumps were being boosted by the RSS pumps. The team verified that during testing of this modification, the correct throttle positions would be set to provide the required system resistances on the basis of observed flows, and if necessary, orifice plates would be resized. The flow model would subsequently be adjusted to conform to the observed system flow conditions.

(b) Procedure EN 311121, "IST Pump Operational Readiness Evaluation," Rev. 6, dated December 10, 1996. (Provides instructions to establish the baseline reference values to analyze data from all pumps tested in the plant as a part of the ASME Section XI testing program.)

Step 4.1.5.a provided conversion factors if the medium being pumped was seawater. This conversion factor was incorrect because pressure in feet of water is a standard term of measurement, expressed in feet of standard water (i.e., fresh water at 68°F), regardless of the density or temperature of the medium being pumped.

Attachment 4, page 22, the TS reference for the containment recirculation pumps curve should have also included TS 4.5.2.f.4.

Attachment 4, page 23, the last pump referenced should have been 3CHS*P3C and not 3CHS*P3A.

Attachment 4, page 24, the pumps identified on this curve were the charging pumps; they should have been the safety injection pumps.

Attachment 4, page 25, the correct TS reference is 4.5.2.f.3 not 4.5.2.f.2.

Attachment 4, page 26, the correct TS reference is 4.6.2.1.b not 4.6.2.1.6.

Allowance was not made for the 10 gpm imbalance that could occur between the injection lines as required by the Westinghouse accident analysis. This would raise the acceptance criteria from $\geq$ 452 gpm to $\geq$ 462 gpm. The team reviewed the most recent test results for the "A" and "C" pumps to ascertain the potential effect of this finding on the current plant condition. They were found to be well above the minimum TS requirement. It was, therefore, considered unlikely that with consideration of the instrument uncertainty, they would have failed to meet this requirement.

The licensee stated that the above minor issues will be evaluated and corrective actions implemented.

2.2 Recirculation Spray System

2.2.1 Scope of Review

Verify that the recirculation spray system (RSS), a support system for the charging system, could perform the required support functions of providing a suction source and pressure boost to the charging pumps during the recirculation phase of an accident.

2.2.2 Findings

The team reviewed licensee event report (LER) 89-012, "Containment Leakage in Excess of Limits Due to Valve Leakage," dated July 15, 1989, which documents the failure of an 10 CFR Part 50, Appendix J, Type C, test of the “A” RSS pump containment sump suction, containment isolation valve, 3RSS-MOV-23A. This valve was a 12-inch butterfly valve with a "vulcanized rubber seat" that had separated from its mounting surface, and as a result, its leakage rate exceeded the total allowable for all containment valves.

Corrective actions included the removal of the valve from the system and shipping it to Pratt, the manufacturer, for overhaul and a post-maintenance local leak rate testing (LLRT) after reinstallation. The team determined from the information provided in the LER, that the valve seat material was not changed. However, the team noted that the valve seating material used may not be correct for this application in that the licensee could not demonstrate that its was adequate for the environment to which it would be subjected.

The LER described the seat material as "vulcanized rubber," a generic term applicable to a myriad of elastomer materials. Per Calculation UR(B)-393-0, "Lifetime Radiation dose to 3RSS*MOV20A, B, C, and D," Rev. 0, dated April 30, 1984, the accident radiation dose to these seats was $2.2x10^7$ rads. Another calculation, UR(B)-400, "Gamma and Beta Equipment Qualification Doses for Normal Operations, Depressurized LOCA and Pressurized LOCA," Rev. 0, predicted the accident dose at $2.4x10^7$ rads. Neither calculation accounted for the 40-year lifetime dose to the valves. (It should be noted that the accident exposure would be concurrent with the accident design temperatures, maximum 280°F and exposure to borated water.) These valves would be required to withstand the 40-year lifetime dose plus the accident radiation dose, temperature, and chemical environment and still be capable of performing their containment isolation function.

The team determined that the valve seats were made of ethylene propylene terpolymer (EPT). Per Parker O-Ring Handbook, ORD 5700, page A2-12., no elastomer compounds, including ethylene propylene compounds, should be considered for exposures above 10^7 rads at room temperature conditions because of their loss of “memory,” and, hence, loss of their sealing capability. For higher temperatures, the document stated that the combination of radiation and temperature would further degrade the materials.

A Parker Seal Company test report that was provided to the team by the licensee showed that the best EPT compounds, S604-70 and E740-70, had compression sets of 20.0 percent and 28.6 percent at 10^7 rads respectively, and were the only compounds that could be recommended for service at 10^7 rads or slightly above for room temperature applications. The Parker Seal test report also indicated that other EPT compounds had a relatively low radiation resistance. To assure that an EPT was qualified, the particular compound would need to be known.

A Pratt Valve Company letter claimed that the material was qualified to 10^8 rads. However, the backup material provided with this letter addressed the material properties of tensile strength, elongation, and hardness but failed to address the compression set. Compression set is particularly important for the seals in question because, unlike static O-ring seals trapped in a seal groove that will fill the space it is trapped in, even if it completely disintegrates, these seals are not confined and the deteriorating material could rapidly fail to provide adequate sealing capability. This backup information also showed some very dramatic degradations in elongation and hardness at 10^8 rads with no indications of what were the maximum allowable degradations. The Pratt Valve Company vendor documents did not identify the specific compounds of the EPT, therefore, the team could not determine the qualification and acceptability for these valve seats.

A significant contributor to the post-LOCA offsite and control room doses is liquid leakage from systems outside of containment. For a single passive failure of a pressure retaining component in the RSS, such as a pump seal, the system's containment isolation valves are intended to limit this leakage to the values used in the dose analyses. If, however, these valves fail to seal because of inadequate design, then the leakage would not be limited, and the offsite and control room doses could exceed the 10 CFR Part 100 and general design criterion/criteria (GDC) 19 limits, respectively.

The team concluded that the RSS pump suction containment isolation valves, 3 RSS-MOV-23A, B, C, and D, had not been demonstrated to be able to perform their design basis function of primary containment isolation under all required conditions. This finding questions the qualification of other similar containment isolation valves. The licensee was asked to provide a list of these valves.

The adequacy of the qualification of the RSS sump isolation valves remains unresolved pending NRC review of the licensee's further analysis related to the qualification of the valves. (URI 423/97-206-05)

2.3 Main Steam System

2.3.1 Scope of Review

The review of the main steam system was not within the originally intended scope of this inspection. However, during the initial document retrieval phase of the inspection, the team was provided a calculation list containing Calculation 313, "Temperature Response of a Motor to the Peak Main Steam Line Break (MSLB) Transient Within the Main Steam Valve Block (MSVB)," Rev. 2, dated August 19, 1993. Because it was the intent of the team to review equipment qualification, this calculation was requested and evaluated.

2.3.2 Finding

Main steam system atmospheric dump valves 3MSS*PV20A-D were provided on each of the main steam lines to allow for automatic relief of overpressure conditions by dumping steam to the atmosphere. In parallel with the atmospheric dump valves, remote manually operated bypass dump valves 3MSS*MOV74A-D were also installed to allow operators to dump steam in a controlled manner during plant cooldown when the nonsafety-related main condenser was not available. Upstream of both valves, on a common line, normally open remote manually actuated isolation valves 3MM*MOV18A-D were designed to meet single failure criteria by isolating the steam line following a failure of the atmospheric or remote operated dump valves thus preventing uncontrolled plant cooldown during or following a plant transient. Both valves are electric motor-operated valves.

In 1994, the licensee performed a modification under DCN NME-S-0091-94, "Electrical Equipment Qualification Master List Deletion of 3MSS*MOV18A, -18B, -18C, -18D (Limitorque Operators)," dated February 25, 1994, that removed the electric motors associated with the isolation valves from the environmental qualification (EQ) list. The licensee's two justifications for this modification were that (1) these valves performed no safety-related function and (2) that single failure was not required to be considered for components in the penetration area. Subsequent to this modification, thermal insulation blankets that protected the valve motors from the steam line break environment were removed.

10 CFR 50.49, Environmental qualification of electrical equipment important to safety for nuclear plants, requires that equipment important to safety be included in a program for qualifying electrical equipment. Included in this requirement are safety and nonsafety-related electrical equipment whose failure under postulated environmental conditions could prevent the accomplishment of safety functions such as controlled cooldown. The team concluded that given the function performed by the isolation valves and their potential exposure to the harsh environment of an accident, the valve operators were required to be qualified and incorporated into the EQ program. Failure to maintain the isolation valves' valve operators in the environmental qualification program is a violation of 10 CFR 50.49 requirements. (VIO 423/97-206-06)

2.4 Conclusions

Among the more significant findings in the mechanical area was the removal of the motor-operated isolation valves for the main steam atmospheric dump and bypass valves from the environmental qualification program. The failure to demonstrate the qualification of the seat material for the RSS containment isolation valves is also a potentially significant issue. The control of calculations is a programmatic issue, which the licensee, as well as the team, identified as a concern. The team noted the document control organization's initiatives to address this issue.

3.0 Electrical

3.1 Onsite ac Source - Emergency Diesel Generator

3.1.1 Scope of Review

The team reviewed from the emergency diesel generators (EDG) A to 4.16 kv bus 34C, down to the 4.16 kv bus 34C and connections, to charging pump 3A.

3.1.2 Findings

An intentional time delay of 0.2 seconds was introduced in the differential protection to allow the clearing of ground faults by the neutral breaker (see calculation 12179GM-60-03.421CB, dated June 1, 1984). The timer (device 1-62G) setpoint calculation did not include any evidence to substantiate the selection of the 0.2 second time delay. The team was concerned about the actions introduced to delay the differential protection. This would have the potential to result in heavier damage to the generator in the case of a full, three-phase, short-circuit, fault in the windings. In view of the fact that one of the principal advantages of the differential protection is its inherent ability for fast detection of faults, the team was concerned that there was no objective evidence that an analysis was performed to provide justification for allowing the differential protection to trip in a longer time than that of its intended design.

On the basis of input information provided by the licensee, the delay in the differential protection actuation would only occur for the case of fault not involving ground (i.e., phase-to-phase or three-phase faults). For a fault involving ground, the neutral breaker would be tripped without delay, thereby, ensuring that the ground fault was eliminated, and, thereby, continuing the operation of the generator. However, for phase-to-phase faults and for three-phase faults, the delay would be in effect, which would subject the machine winding to the potential destructive thermal and mechanical effects of very high and prolonged short-circuit currents. Because the other train would be available to provide required emergency power, the team considered this an equipment protection issue rather than a safety issue. The licensee initiated an evaluation to determine the acceptability of the of the timer setting of 0.2 seconds.

3.2 Medium Voltage Distribution and Medium Voltage Switchgear

3.2.1 Scope of Review

Medium voltage distribution and medium voltage switchgear was evaluated from the 4.16 kv bus 34C through 480 V substation 32R, 480 V MCCs 3A2 and MCC 3A1, to charging pump and ventilation system valves and motors.

3.2.2 Findings

(a) Surge protection of medium voltage equipment

One-Line Diagram 12179-EE-1C, indicated that surge arresters were provided on the 345 kv side of the RSST. The team requested an evaluation of the protection afforded by these arresters for surges transmitted through the transformer windings by capacitive and inductive effects. The licensee performed a study in response to the team's question. The evaluation considered as acceptance criterion the impulse withstand capability of the medium voltage switchgear and not the impulse withstand capability of the class 1E motors connected to the bus. The impulse withstand of the motors would tend to be quite lower than that of the switchgear, due to the limited insulation and geometric distance between the winding turns. Therefore, use of the impulse withstand capability of the switchgear as acceptance criterion was judged to be nonconservative because of the potential common mode failure constituted by a single surge traveling down the transformer and causing simultaneous damage to redundant class 1E motors of the two safety-related trains.

The licensee indicated that the evaluation of the surge protection will continue. Adequate surge protection for 1E motors is considered an unresolved item pending the licensee's evaluation of the plant's surge protection. (URI 423/97-206-07)

(b) Protection of cables against short circuits in the end load devices

The team was unable to find any evidence in Specification SP-M3-EE-269, "Electrical Design Criteria", Rev. 1, dated July 23, 1997, that cable insulation had to be designed to survive the short circuit currents flowing to faults in downstream parts of the distribution system. The team found this issue significant because the plant committed to adherence to IEEE Standard 308. Paragraph 5.2.1(6) of that standard, "Protective Devices" states "Protective devices shall be provided to limit the degradation of the Class IE power systems," implying that only the part of the system where the fault has occurred should be expected to become damaged. In cases of cables feeding loads, IEEE 308 dictates that the cable should not be damaged for faults occurring at the load device. Faults at the terminals of load devices have a higher probability than faults in the cable itself. Therefore, the team requested information on how many other safety-related cables could sustain damage in case of faults at the terminals of the load devices. Upon further investigation, the team found that a number of the safety-related 4 kv cables for Millstone Unit 3 could be damaged by faults at the load devices. The safety significance of this issue is mitigated by the fact that the consequences of an assumed electrical fault would affect only one of the two redundant electrical divisions.

The licensee indicated that the short circuit current levels were in the process of being re-evaluated, and that this issue was a start up issue. (Ref. CR M3-97-2358, dated February 25, 1997). The evaluation of the qualification of unit cables to withstand short circuits is an Inspection Followup Item pending the licensee's full identification of the problem and NRC review of the licensee's evaluation. (IFI 423/97-206-08)

3.3 125 Vdc Batteries

3.3.1 Scope of Review

The system was evaluated from the 480 V MCC 3A2, down to the battery charger 301A-2, to 125 Vdc battery 1, down to 125 Vdc Panel 3BYS*PNL-1, to 3BYS*PNL-1V, to fuse distribution panel 3BYS*PNL17F and to 125 dc loads associated with systems inspected, also from the 480 V MCC 3A2, down to inverter INV-3, and the 120 Vac loads, and from the 125 Vdc distribution panel 301A-2, down to inverter INV-3, and the 120 Vac loads.

3.3.2 Findings

(a) Calculation of battery capacity

The battery capacity Calculation BAT-SYST-1240E3, Rev. 1, did not consider the condition of plant operation involving safety injection (SI). Calculation BAT-SYST-1240E3, Rev.1, paragraph 6.1, "Scenario Development," page 21, contains conclusions from quoted paragraphs of the FSAR that "the design basis event (DBE) is a Loss of Power (LOP) with loss of battery charger." A further statement is made to the effect that SI does not have to be considered as a design condition. "SI, concurrent with a LOP and loss of battery charger output, is not part of the design basis for the two hour battery discharge." While it may be acceptable to disregard SI conditions for the battery 2-hour rating, it is not acceptable to do so for the 1-minute rating of the battery (Ref. Criterion 17, and FSAR, paragraph. 8.3.2.1). This is because the 1-minute rating would apply under a loss-of-offsite power (LOOP) and SI conditions for the period of time until the EDG can be started and connected to the bus, and is able to provide power to the battery charger.

The licensee indicated that the SI conditions were enveloped by the analysis conducted and that a clarification would be provided to the calculations. This is an unresolved item pending NRC review of the licensee's clarification of the battery calculations.
(URI 423/97-206-09)

(b) Lack of labeling on battery room hydrogen concentration indicators

The hydrogen analyzers instrumentation did not indicate the units of measurement nor the identification of the battery in which they were associated. The lack of this information could result in noncompliance with Station Procedure SP 3712 NA, "Battery Surveillance Testing," precaution 3.1.5, which states "Read hydrogen meter prior to entry into any battery room; maximum concentration for entry is 2 percent hydrogen."

The licensee agreed that adequate labeling would be required, and initiated the corrective measures by Station Procedure OA9, Attachment 2, "Label Request Form," to develop a label for each Hydrogen Analyzer to indicate its ID, the unit of measurement and its associated battery. This is an Inspection Followup Item. (IFI 423/97-206-10)

3.4 Degraded grid protection

3.4.1 Scope of Review

Calculations, assumptions, and protective device settings associated with the protection of plant equipment from the effects of low voltage on the offsite grid were evaluated.

3.4.2 Findings

(a) Time Delay for Undervoltage Protection

The team reviewed Calculation NL-040, and found insufficient substantiation for the selection of a long, 1.8-second delay, in the output of the undervoltage protection, which, as defined, operates below 70 percent voltage. There was no evidence that the selection of the time delay had been examined in light of the related consequences of allowing plant operation at zero voltage. If the undervoltage condition were such as to cause total voltage collapse (a level of zero volts), all 480 V magnetically held contractors would drop out, and all running 4.16 kv and 480 V motors fed from circuit breakers would slow down, with many stalling due to the collapse in input power. The drop out of contactors would cause associated safety-related loads to be disconnected. In the unlikely event that voltage restoration occurs within 1.8 seconds prior to the under voltage relay completing its protective function, simultaneous motor reaccelerating and restarting would occur, which may cause over currents sufficient to produce undesired tripping of protective relays. The team did not find justification for the assumption that a 1.8-second delay would be acceptable in the case of total voltage collapse. The licensee stated that the delay setting of 1.8-second will be evaluated. This is an unresolved item pending NRC review of the licensee's evaluation of the 1.8-second delay. (URI 423/97-206-11)

3.5 Conclusions

There were no violations identified. Calculation inconsistencies and lack of adequate document bases for calculations, setpoints and conclusions will be addressed by the licensee.

4.0 Instrumentation and Control

4.1 Transfer from Normal Chemical and Volume Control System Lineup to High-Head Safety Injection and Containment Sump Recirculation

4.1.1 Scope of Review

Evaluate instrumentation and controls associated with the transfer from normal CCS lineups to high-head safety injection and from high-head safety injection to high-head containment recirculation were evaluated. Documentation reviewed included TSs, P&IDs, elementary wiring diagrams, logic diagrams, test loop diagrams, instrument uncertainty calculations, hydraulic calculations, calibration procedures, calibration results, and normal and emergency operating procedures. Equipment walkdowns were conducted in the main control room (MCR), auxiliary shutdown panels (ASP), charging room cubicles, charging pump cooling heat exchanger area, and along the charging and high-head safety injection flow paths.

4.1.2 Findings

(a) Improper description of valve logic

Valves 3CHS*8804A and 3CHS*8804B open during emergency core cooling system (ECCS) suction transfer from the refueling water storage tank (RWST) to the containment sump. The control logic for the valves is very similar. However, FSAR Table 6.3.3 "Motor Operated Valves in the ECCS System," had incorrect logic information for these valves. This was previously identified by the licensee and was addressed under FSAR Change Request (FSARCR) 97-MP3-323. The team noted that the interlock description for valves 3CHS*8804A and 3CHS*8804B were different. 3CHS*8804A was correct, but 3CHS*8804B was still incorrect. The draft FSARCR stated 3CHS*8804B cannot be opened unless residual heat removal (RHR) inlet isolation valve 3SIH*8702A, or 3SIH*8702B, or 3SIH*8702C is fully open. The correct interlock description should be that 3CHS*8804B cannot be opened unless 3SIH*8702A, or 3SIH*8702B, or 3SIH*8702C is fully shut.

The failure to properly correct a deficiency in the FSAR is an example of a violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. (VIO 423/97-206-12)

(b) Inadequate Valve Test Procedure

Procedure SP 3604A.5, Rev. 10, "Chemical and Volume Control System Valve Operability Test," includes testing of alternate charging path valves 3CHS*190A and 3CHS*190B. These valves should only be tested during cold shutdown conditions when normal charging is isolated, but this prerequisite was not included in the procedure. The operations procedures (OPs) group initiated a Station Administration Procedures Group Report Feedback Form to add this prerequisite to the procedure.

The failure to correctly specify when to test CHS*190 A and B is an example of a violation of Technical Specification 6.8.1, inadequate procedures. (VIO 423/97-206-13)

(c) Inadequate bypassed and inoperable status indication

FSAR discrepancies in Section 7.3.1.1.5 with bypassed and inoperable status indication equipment was being corrected under FSARCR 97-MP3-101. However, the approved FSARCR did not identify that the SI pumps cooling pump being inoperable caused the SI pump itself to be inoperable. CR M3-97-3134 was written to correct the FSAR.

The failure to properly correct a deficiency in the FSAR is another example of the apparent violation of 10 CFR 50, Appendix B, Criterion XVI, Corrective Action. (VIO 423/97-206-12)

(d) Improperly labeled auxiliary shutdown panel pushbuttons

Auxiliary shutdown panel (ASP) pushbuttons for charging pump suction valves 3CHS∗112B and 112D were labeled "OPEN/AUTO" and "CLOSE/AUTO," and the control switch for SI pump 3SIH∗P1A was labeled "STOP-AUTO-START." These Train A devices do not have an automatic function from the ASP because of safe shutdown concerns, although their corresponding B train components in the ASP and both trains in the MCR do have automatic functions. Resolution to this labeling problem is being tracked under CR M3-97-3160.

(e) Improper RWST draw-down assumption

Calculation 12179-US(B)-295 Rev. 5, with Change Notice 1, "RWST Draw-down Rates and Switch Over Levels," demonstrates that 20 minutes of ECCS flow is available prior to the manual transfer from cold-leg injection to cold-leg recirculation modes. The maximum charging pump flow assumed in the this calculation and the FSAR was 820 gpm. Analysis for the Vantage 5 fuel in Westinghouse Letter No. FSSE/CWBS-1200 "MP3 FSAR Input Data," indicates charging flow could be 876 gpm. Therefore, the worst-case drawdown time would be marginally less than that indicated in the calculation. This is addressed in CR M3-97-2956.

4.2 Refueling Water Service Tank

4.2.1 Scope of Review

The team reviewed instrumentation and controls associated with the refueling water service tank (RWST) 3QSS*TK1 including elementary wiring diagrams, logic diagrams, test loop diagrams, isometric drawings, instrument setpoint calculations, calculations instrument uncertainty for hydraulic calculations, calibration procedures, calibration results, and emergency operating procedures. Walkdown and verification of safety-related level transmitters and switches were also conducted.

4.2.2 Findings:

TS 3.5.4.a requires at least 1,166,000 gallons in the RWST which corresponds to approximately 57 and 59 feet of water in the tank. An overflow pipe limits the tank volume to 1,186,110 gallons, resulting in 20,110 gallon TS control band.

Surveillance Test Procedure 3604C.1 is used to verify compliance with TS 3.5.4.a. that requires the RWST water level to be maintained above the minimum level of 1,166,000 gallons. The procedure specifies the use of the wide range instruments for establishing RWST levels.

A document identified as WCAP-14353, "Westinghouse Setpoint Methodology for Indication, Control, and Protection Systems for Millstone Nuclear Power Station Unit 3 - 24-Month Fuel Cycle Evaluation," Rev. O, Tables 3-109 and 3-110, indicates that the wide-range RWST-level instruments have an uncertainty of 5.1 percent of span for the analog control room indicators, and a 4.2 percent of span for the computer points corresponding to RWST volumes of 61,557 and 50,594 gallons, respectively. Since the margin of instrument error is greater than the specified control band, surveillance test procedure 3604C.1 cannot verify the compliance with TS 3.5.4.a.

The failure of ST 3604C.1 to ensure the TS requirement is another example of a violation of Technical Specification 6.8.1, inadequate procedures. (VIO 423/97-206-13) The team verified that existing control room annunciators which are actuated by a more accurate narrow range instrument could be used by control room operators to obtain accurate RWST level.

4.3 Calculation Content and Control

4.3.1 Inspection Scope:

The team reviewed several calculations as well as procedures describing how instrument uncertainty calculations are performed.

4.3.2. Findings

(a) Errors in RWST setpoint calculation

There were several minor errors in RWST setpoint calculations dating from the original design period as well as in new calculations. Setpoint calculation SP-3QSS-1, Rev. 1, establishes the voltage level associated with the RWST high temperature alarm. The desired setpoint is 49°F, but because of an error in the calculation, the actual specified setpoint is 3.2 volts which equates to 48°F. The associated Loop Calibration Report indicates that the instrument was calibrated to the desired setpoint of 49°F, and not the incorrect value in the calculation. Therefore, there are two errors: (1) the calculation is wrong, and (2) the plant procedures do not agree with the calculation. This resulted in action request (AR) 97021209 being written to revise the calculation. Additionally, setpoint calculations SP-3QSS-5, Rev. 1, and SP-3QSS-6, "3QSS-TS38 RWST Temperature," Rev. 0, establish setpoints for RWST temperature control. They incorrectly refer to valves 3CDS-SOV26A and 26B. Only one valve 3CDS-SOV26 exists. CR M3-97-2797 was initiated to resolve the problem.

(b) Errors in the RWST level interlock channel calibration

Calculation 3451B03-1232 E3, Rev. 0, "RWST Level Interlock Channel Calibration," is a new calculation; it was performed using a spreadsheet and verified using hand calculations. It provides values used to calibrate the RWST low-low and tank empty level switches. The team identified the following concerns:

- The use of an incorrect instrument uncertainty value for tank empty level switches 3QSS*LS56A-D was a result of inadequately justified assumptions for boron concentration, temperature, and tank empty level switch drift values . Equations used in the numerical calculations were not included in the calculation used in the Microsoft Excel® spreadsheet which was not maintained as a QA document.

- The use of incorrect instrument drift values could result in the instruments being inaccurate and that not being noted during their associated surveillance test procedures.

Condition Report M3-97-3169 was written to address some of these problems.

The followup of 4.3.2a and b is identified as an Inspector Followup Item (IFI 423/97-206-14).

4.4 Conclusions

The team identified a number of issues regarding instrument uncertainty. Adequate consideration of instrument uncertainties was raised on Unit 2 in adverse condition report (ACR) 3577. The resolution of that ACR should be broadened in order to address the concerns raised in this area as well as those in the mechanical portion of this inspection.

5.0 Structures and Supports

5.1 System Modifications

5.1.1 Scope of Review

System modifications including the modification that was required to install the new restrictive orifices 3SIH*R038, 039, 041, were evaluated. These orifices were installed to protect the hot and cold-leg injection valves from erosion.

5.1.2 Findings

Calculation Change Notice (CCN) No. 6, was performed to justify the installation of restrictive orifices 3SIH*R038, 039, and 041. The installation of these orifices is accomplished by removing a

14-inch section of a 1.5-inch schedule 160 pipe and installing a restrictive orifice with an outside diameter of 3.0 inches and an inside diameter of 0.3 inches.

The piping calculation that was performed to substantiate the acceptability of CCN No. 6 to Calculation 12179-NP(B)-X100700, Rev. 1, did not include the stress indices that are required for ASME Class 1 piping analysis in accordance with NB-3600 of the ASME Code.

The existing calculation revision (through CCN 5) had no restrictive orifices. CCN No. 6 considered only the effects that the increased weight/mass (of the new orifices) would have on the system response, stresses, and pipe support loadings on adjacent supports. However, the calculation failed to provide consideration for the different stress indices as required by NB-3600. Since the indices for the restrictive orifices are higher than for straight pipe, the resulting stresses and interactions would also be higher. Therefore, the calculation should have provided considerations for these higher indices.

The safety significance of this finding appears to be minimal. The restrictive orifices were installed in a section of ASME Code Class 1 piping which experiences relatively mild transients. The resulting stresses, although higher, will still be less than the allowable stress limits. The failure to consider the stress indices required by the ASME Code is another example of a violation of 10 CFR 50.55a. (VIO 423/97-206-03)

5.2 Piping Stress Calculations

5.2.1 Scope of Review

Review analysis procedures and guidelines that were used to perform the piping and pipe support analysis include M149 - Specification for Piping Engineering and Design ASME III; Code Calculation 1, 2, 3; ANSI B31.1, Class 4; and Stone & Webster Engineering Corporation, North East Technical Memorandum (NETM) NETM-30 Procedure for Preparation, Review, Approval and Control of Power, Hydraulic, and Engineering Division Stress Data Packages and Engineering Mechanics Division Documents.

Large bore piping including suction piping from RWST to charging pumps was reviewed. This Section included yard piping (above and below ground), piping runs in engineered safety feature (ESF) building, pipe tunnel, and auxiliary building. Piping analysis was reviewed including different amplified response spectra, buried piping, and outside piping.

Additional large bore piping from charging pump discharge to the RCS injection nozzle for both the injection and recirculation modes were reviewed. Features reviewed included the containment penetration anchor and the ASME Code Class 1 and 2 parts of the charging pump discharge.

The following features were also evaluated:

- small bore piping stress analysis providing cooling to the charging pumps bearings
- charging pump calculations for pump foundation and nozzle loading
- approximately 15 pipe support calculations (both large and small bore)
- containment unsleeved piping penetration calculation for charging pump discharge
- RHR line rupture restraints and associated space frame

- steel load reverification program and analysis for containment annulus pipe rack framing

5.2.2 Findings

(a) Response spectra analysis

The team identified a potential source of nonconservative analysis. The procedures require that only the response from frequencies under 50-Hz or from the first 50 modes (whichever comes first) need to be considered. In the case of a rigid system (i.e., significant frequencies greater than 50-Hz) or a very flexible system (i.e., many more significant modes than 50-Hz) potential exists that not all or most of the mass will be considered in the analysis.

The FSAR states: "All significant dynamic modes of responses under seismic excitation with frequencies less than 50-Hz or modes less than 50, whichever is reached first, are included in the dynamic analysis described in FSAR Section 3.7B.3.8." The concern is that these cutoff frequencies do not ensure that all or most of the system mass is accounted for in the seismic analysis. The three areas of particular interest are: (1) stiff systems where significant parts of the response are above the 50-Hz cutoff, (2) relatively large and soft systems where there are numerous modes less than 50-Hz such that the 50-Hz modes cutoff would not include most of the system mass, and (3) long piping runs with an axial restraint (i.e., long rigid runs greater than 50-Hz). For these types of systems the pipe support loadings may be under predicted.

The licensee indicated that they would review the calculation and evaluate the team's finding.

This is an unresolved item pending licensee analysis of pipe support loading.
(URI 423/97-206-15)

(b) Masses not considered in analyzing containment annulus pipe rack structures

The team identified a concern that the entire mass of the attached piping was not considered in the analysis of the rack structure. The rack structure was considered rigid without any justification for the assumption. Therefore, the rigid range accelerations were used in the rack calculation. The actual frequency of the structure could be significantly less than assumed when all the attached pipe support mass is added to the mass of the structure. A lower frequency would result in a higher structural amplification of the acceleration.

The annulus pipe rack calculation demonstrates that the rack is structurally adequate to withstand the revised pipe support loadings generated during the 1996, Piping Reconciliation Effort because of an elevated temperature of the SI system and other piping systems attached to the rack structure. The main function of the pipe rack structure is to provide a structure to support piping, conduit and ducts in the annulus area. Therefore, the most significant loadings on the structure are the forces and moments as a result of supports for piping, conduit, and ducts. A review of the calculation revealed that the structure was accelerated using the rigid range accelerations (i.e., the structure was assumed rigid). The calculation provided no justifying calculation or explanation as to why the structure could be considered rigid. This structure, unlike a normal pipe support, is a very large frame structure with numerous attached concentrated masses (i.e., supported piping). It appears to be prudent to consider the potential amplification of the structural system when performing the structural verification calculation. The inclusion of attached masses and the resulting higher structural amplification should be considered as they will increase the total loading and stresses for the rack structure should be considered.

The safety significance of this finding for the specific area appears to be minimal since the calculated stress for the annulus pipe rack indicates that a relatively large margin exists using the current loadings. The application of the larger amplified structural response is not expected to result in stresses or loadings that exceed the current allowable limit.

This is a violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control.
(VIO 423/97-206-16)

5.3 Thermal Expansion of Structural Steel

5.3.1 Scope of Review

Evaluate the effects of the thermal expansion of steel structures that were not considered for the elevated containment temperature following an accident. Post accident temperature of the containment can be as high as 280 ^{o}F causing the structural steel to expand. That expansion will cause thermal loadings if the structure is configured such that the thermal expansion will be resisted.

5.3.2 Findings

The team identified at least one structure (pipe whip structural in the steam generator cubicle area) that has the potential to generate significant loadings as a result of the restraint of the structural steel growth.

The structural steel reviewed has been essentially installed to provide support for pipe whip restraints. The main members consist of W14x343 members which are rigidly framed between the crane rail wall and the inner cubicle wall. The concern is the effect of the differential in thermal expansion between the concrete and steel when the containment heats up to 280 ^{o}F. The linear coefficient of expansion for steel and concrete are about the same (6.6×10^{-6} for steel vs 5.5×10^{-6} for concrete). Therefore, under steady state conditions, there will be negligible stress or loading because of differential thermal expansion. However, under transient conditions, the steel will absorb heat at a significantly higher rate than the much more massive concrete structure resulting in stresses as a result of end constraint.

The thermal growth will be resisted by the relatively rigid walls. The maximum generated force exerted on the steel and wall will be approximately 4,000 Kips. This force will be relieved by any clearances that may be available because of bolted joints (after slippage), and the available structural flexibility of the wall and steel structure. The resulting loadings will need to be checked against the wall/structural capacity to determine the severity of this concern. In addition, consideration must be given to assessing the effects of these loadings in conjunction with other loadings that may potentially be concurrent with or follow this event.

The safety significance of this finding is indeterminate at this time for this structure or other similar structures. This item is considered unresolved pending licensee identification of all structures where this may be a problem and NRC review of the licensee's conclusions.
(URI 423/97-206-17)

5.4 Conclusions

The principal finding of the team in this area was the inadequate consideration of the effects of accident temperatures on rigidly restrained steel.

6.0 Operations

6.1 Charging and Safety Injection System Operating Procedures

6.1.1 Scope of Review

Evaluate the charging and safety injection system emergency, abnormal and normal operating procedures, TSs, Technical Requirements Manual (TRM), FSAR description, and accident analysis. Observe operators perform surveillance procedures.

6.1.2 Findings

(a) Inadequate Technical Specification 3.4.3

Millstone Unit 3 TS 3.4.3 states, in part, "The pressurizer shall be operable with a water volume of less than or equal to 92 percent (1656 cubic feet)." The bases is that TS states, in part, "The limit is consistent with initial SAR assumptions." However, FSAR Figure 15.5-2, which documents the pressurizer level transient resulting from an inadvertent ECCS injection at full power, starts at approximately 1,250 cubic feet (the upper level of the normal pressurizer control band).

On the basis of the transient from which that figure was derived, FSAR Section 15.5.1.3, concludes that following an inadvertent ECCS actuation, "the pressurizer will not reach a water-solid condition with or without power operated relief valve (PORV) actuation, prior to 10 minutes from event initiation." (It is assumed that no operator action will be taken for 10 minutes after initiation as the operator goes through the appropriate procedures to determine whether the ECCS actuation was a valid or invalid signal, in which case injection should be terminated.)

In reviewing this transient, which would result from charging pump injection, the team identified an apparent inconsistency between the TS and the conclusion of the FSAR. If the transient described in the FSAR was initiated at the highest TS allowable pressurizer level rather than at the top of the normal control band, it appears likely the acceptance criteria of not reaching a solid-water condition within 10 minutes could not be achieved. When this issue was raised with the licensee, the team was informed that this issue had been previously identified in CR M3-97-3228.

In dispositioning the CR, the licensee concluded on the basis of information from Westinghouse, that the wording of the TS was never meant to imply that the limiting condition for operation established the initial condition for accident analysis and that Westinghouse had submitted a change request to the standard Westinghouse TS (NUREG-1431) to delete the statement discussed above from the TS bases. However, the team noted that it was not just the TS bases that would appear to imply the use of the higher pressurizer level in the analysis. FSAR section 15.0.7 states, in part, "control system action is considered only if that action results in more severe accident conditions. No credit is taken for control system operation if that operation mitigates the results of an accident."

Even assuming a failure of the pressurizer level control system, it is extremely unlikely that the pressurizer would ever be operated at or near the TS pressurizer high-level limit for any significant length of time. The pressurizer high-level annunciator alarm at 70 percent and the operator response to the corresponding procedure would assure that under all likely conditions an inadvertent ECCS actuation would occur at or below a pressurizer level which would ensure the FSAR acceptance criteria would be met. Nevertheless, given the language of the FSAR, Section 15.0.7, the licensee’s CR conclusions appear to be inconsistent with the FSAR’s description of the accident analysis methodology.

The licensee’s proposed change to the TS to resolve the apparent inconsistency with the FSAR will be evaluated by the NRC. This is an Inspector Followup Item. (IFI 423/97-206-18)

(b) Technical Specification 3.7.14, "Area Temperature Monitoring," is inadequate

During the inspection of the ventilation that supports operation of the charging and charging cooling pumps, the team noted that the licensee has previously identified a number of problems with calculations and procedures associated with the system. For example, the licensee identified that the charging pump area temperature alarms were set at 119 ^{o}F (which is above the TS limit of 110 ^{o}F) and that design calculations show that temperatures in the those areas could reach 111 to 113 ^{o}F, under normal and accident conditions respectively, when the design temperature was 104 ^{o}F.

After reviewing the TS and the licensee’s responses as to why the various design, environmental qualification, and operational temperature limits are consistent (with the above noted exceptions), the team noted the following.

First, the APPLICABILITY statement of the TS is improper. The TS is only applicable when the equipment is required to be operable as stated in the TS. However, the basis of the TS is environmental qualification and not operability. Therefore, the TS should be applicable at all times given that elevated temperatures in the rooms can affect equipment qualification whether or not it is required to be operable. Second, the ACTION statement a. of TS 3.7.14 is inadequate. The ACTION statement only requires recording the cumulative amount of time by which the temperature exceeds the limit if that is less than 20 ^{o}F and for less than 8 hours. Because the maximum abnormal excursion (MNE) temperature (a one-time, 8-hour, maximum temperature the equipment is postulated to experience before an accident) for the charging pump room, for example, is 120 ^{o}F (while the TS limit for that room is 110 °F), the ACTION statement fails to ensure that the charging pumps will be maintained at or below required environmental temperatures or that such exposures will be sufficiently limited during the life of the equipment. Additionally, the licensee informed the team that the temperature value used in the TS is equal to that of the MNE. If the TS temperatures are set exactly at MNE, it is unclear how the 2.2 ^{o}F instrument error discussed in the TS is accounted for in the limit. Finally, the rationale for having a TS to control plant area high temperatures for EQ purposes and only the controlling operability-related low temperature for the charging pump rooms in the TRM is unclear.

The safety significance of these findings is low. Even with the improper setting of the charging area temperature alarms there are a number of other alarms and indicators that would alert the operator before temperatures in the area would be a problem even under elevated ambient temperature conditions. With regard to the low temperatures presently controlled by the TRM, the action required by the TRM and the definition of system operability contained in the TS should adequately address this issue while the problems related to the TS are being resolved.

The licensee’s corrective actions to resolve the inconsistencies in temperature settings is an Inspector Followup item. (IFI 423/97-206-19)

(c) FSAR description of auxiliary building ventilation inconsistent with design

FSAR Chapter 9.4.3.1, Item 12, states that air flow in the auxiliary building shall be maintained from the least contaminated to the more contaminated spaces. The team noted during the LOCA recirculation mode of operation that the charging pump cubicles would potentially be the most contaminated areas in the auxiliary building. In the ventilation system’s winter alignment, the auxiliary building ventilation system draws air from the charging pump cubicles and distributes a part of the air to other areas in the auxiliary building in lieu of the charcoal filters. At the end of the inspection period, the licensee issued CR M3-97-03161 to resolve the FSAR discrepancy associated with the flow of contaminated air in the auxiliary building.

The inconsistency between the FSAR and the design of the charging pump ventilation is another example of a violation of 10 CFR 50.71 (e). (VIO 423/97-206-01)

6.2 Charging and Safety Injection System Annunciator Response Procedures

6.2.1 Scope of Review

Evaluate applicable emergency operations procedures (EOPs), annunciation response procedures, and OPs.

6.2.2 Findings

(a) Containment recirculation cooler SW flow annunciator response inadequate

During review of the annunciator response procedures, the team identified that OP 3353.MB1C, “Main Board 1C Annunciator Response,” Rev. 1, Change 5, 1-1B “CTMT Recirc CLR SW FLOW HI/LO” was inaccurate.

For Train A, Step 6.2.2 requires that if there is a SW rupture downstream of 3 SWP*MOV57A and 3 SWP*MOV57C refer to OP 3308 "High Pressure Safety Injection and STOP 3 SIH*PIA, safety injection pump.” For Train B, Step 6.4.2 requires similar action but conditions the response with "IF Safety Injection is not actuated."

Given that the equipment in question would only be in operation during an accident or during surveillance, the Train A direction to stop a safety injection pump under all conditions is improper, in that, it directs safety equipment to be secured during an accident. The inconsistency in the annunciator response procedure for the SW flow to the recirculation spray heat exchanger is another example of a violation of Technical Specification 6.8.1, inadequate procedures. (VIO 423/97-206-13)

(b) Reactor plant composite cooling water exchanger SW flow annunciator response inadequate

The alarm noted that setpoint of the low service water flow to the reactor plant component cooling water system (RPCCW) heat exchanger is set at 6200 gpm. The present calculations show that 7125 gpm are needed to achieve safety-grade cold shutdown. The licensee indicated that the setpoint (which is also the service water flow rate to be restored according to abnormal operating procedure (AOP) 3561, "Loss of Reactor Plant Cooling Water," Step 2.b) is not based on the service water flow needs for safety- grade cold shutdown. The AOP valve is based on service water flow needs during accident conditions, with some consideration for setting the alarm low enough to preclude spurious alarms. Verification of adequate service water flow to accomplish safety-grade cold shutdown is provided by a weekly test performed on the service water side of the RPCCW heat exchangers which has an acceptance limit of 7200 gpm.

While having the annunciator and AOP values set on the basis of the safety required accident flow rate and a weekly test to verify flow set on the basis of safety-grade cold shutdown requirements appears reasonable, there is one weakness in the procedural implementation of this arrangement. Should the annunciator actuate or the AOP need to be entered because of low service water flow, there is no specific guidance in the restoration steps of the AOP or in the annunciator response procedure that directs the operator to subsequently verify flow that safety-grade cold shutdown has also been reestablished. As implemented, the annunciator could alarm, flow could increase just enough to clear it, and the unit could remain in a condition without adequate flow to achieve safety-grade cold shutdown until the next surveillance (up to one week).

The inconsistency in the required service water flow between the AOP, the annunciator, and the weekly surveillance test is another example of a violation of Technical Specification 6.8.1, inadequate procedures. (VIO 423/97-206-13)

6.3 Operator Training

6.3.1 Scope of Review

Evaluate the charging and safety injection system operator training and design change training modules associated with charging pump cubicle and charging system valve disk modifications.

6.3.2 Findings

The team found that the training modules adequately addressed system modifications and operations.

6.4 Conclusions

The team found the system engineers assigned to the charging system and other interfacing systems to be strength. The assigned engineers had a thorough understanding of their assigned systems and exhibited questioning attitudes.

7.0 Maintenance

7.1 Charging and Safety Injection System Maintenance Procedures

7.1.1. Scope of Review

Evaluate the charging pump and charging pump cooling pump vendor technical manuals, witness maintenance performed on the charging pump cubicles ventilation duct, and conducted an in-depth system walkdown. Review preventive maintenance, work requests, condition reports, design changes, and maintenance procedures for the following components:

- Charging pumps 3CHS*P3A,B,C
- Charging pump cooling pumps 3CCE*P1A,B
- Charging pump discharge check valves 3CHS*V46,47,48
- Charging pump recirculation valve 3CHS*MV8110
- Charging pump suction valves 3CHS*LCV112B,C,D,E
- Charging pump discharge valves 3CHS*MV8801A,B
- Charging pump cooling heat exchangers 3CCE*E1A,B

7.1.2 Findings

Work Order M3-96-05462 indicated that Teflon tape was used to seal the resistance temperature detector thermowells on the charging pump 3CHS*P3B gear box. The use of Teflon tape was confirmed during the walkdown of Pump 3CHS*P3B. As a result of this finding, the licensee issued CR M3-97-2907 that stated that Teflon is not compatible with the radiation dose rates expected in the charging pump cubicles during the recirculation phase of a loss-of- coolant accident. CR M3-96-0067 had previously identified the use of Teflon tape on components in the containment. Corrective action required that maintenance personnel be notified that the use of Teflon tape was not acceptable and that Teflon tape be remove from the components inside the containment. Corrective action for CR M3-96-0067 did not address the identification of Teflon tape on components located outside of the containment.

The use of Teflon tape on components outside containment is another example of an apparent violation of 10 CFR Part 50, Appendix B, Criterion XVI, for inadequate corrective action. (VIO 423/97-206-12)

7.2 Maintenance-Related Information Notice (IN) Review

7.2.1. Scope of Review

Review licensee's evaluations for the following NRC INs.

- 92-61: Loss of High-Head Safety Injection
- 93-13: Undetected Modification of Flow Characteristics in the High-Pressure Safety Injection System
- 93-42: Failure of Anti-Rotation Keys in Motor-Operated Valves Manufactured by Velan
- 94-59: Accelerated Dealloying of Cast Aluminum-Bronze Valves Caused by Microbiologically Induced Corrosion
- 94-76: Recent Failures of Charging/Safety Injection Pump Shafts

7.2.2. Findings

The team found that the INs were properly evaluated and corrective actions were implemented by the maintenance program when appropriate.

7.3 Conclusions

Overall, based on this inspection, the team found maintenance to be strength as a number of longstanding charging system maintenance problems appear to have been corrected through aggressive action.

8.0 Surveillance

8.1 Technical Specification Surveillance

8.1.1. Scope of Review

Review the surveillance procedures that implement charging/safety injection TS Surveillance Requirements 4.5.2.b, 4.5.2.e.1, 4.5.2.e.2.a, 4.5.2.f.1, 4.5.2.g.2, and 4.5.2.h.

8.1.2 Findings

(a) Air in RSS piping during cold-leg recirculation

During the review of surveillance procedures that vent the charging system ECCS piping, the team found that sections of RSS piping upstream of valves 3RSS*MV8837A, 3RSS*MV8837B, 3RSS*8838A and 3RSS*8838B are maintained in a dry layup condition. Each of these four separate sections of piping are approximately 14 to 17 feet in length and 8 inches in diameter. During the ECCS recirculation mode of operation, two of the four sections of dry piping would be aligned to the suction of the charging and safety injection pumps. The team had two concerns with this configuration. First, there is no test or analysis that verifies that injecting the air in the piping to the suction of the pumps will not unacceptably degrade pump or system performance. Second, the effects of potential water hammer, as water is injected into this dry piping with its significant bends and elevation differences, had not been analyzed.

Licensee evaluation of IN 88-23, Potential for Gas Binding of High Pressure Safety Injection Pumps, including Supplements 1 through 4, did not identify that there was air in the RSS pump discharge and charging and high-pressure safety injection pumps' suction piping. Although the IN and supplements did not specifically address the licensee's configuration, the potential for air and the adverse effect of air in ECCS pumps suction piping was thoroughly discussed. Given that the RSS system configuration is unique, in that it actuates from a dry condition, a thorough review of the system on the basis of the IN should have identified this issue.

Subsequent to the inspection, the licensee informed the team that on August 22, 1985, a Design Deficiency Report (DDR) had been initiated that discussed the potential for air in the recirculation spray system being injected into the suction of the charging and safety injection pumps. After reviewing DDR 641, the team concluded that the DDR focused on another potential air problem (which has subsequenty been resolved) and failed to include the piping of concern.

The failure of the licensee to properly evaluate and correct the potential for the injection of air into the charging and safety injection systems is an apparent violation of 10 CFR 50, Appendix B, Criterion XVI. (EEI 423/97-206-20)

(b) Inadequate venting of RSS piping

TS 4.5.2.b.1 requires that in Modes 1, 2, 3, and 4, ECCS piping, with the exception of the RSS pump, heat exchanger and associated piping, be verified to be full of water by venting the ECCS pump casings and accessible discharge piping high points every 31 days. The team found that the charging pump discharge piping was vented in accordance with this TS requirement. However, one section of RSS pump discharge piping that supplies the suction to the charging and high-pressure safety injection pumps during the recirculation mode of operation was not vented in accordance with this TS requirement. The purpose of procedure SP3610A.3, "RHR System Vent and Valve Lineup Verification," Rev. 3, is to perform this TS requirement, and the procedure did not include vent valve 3SIL*V992. The licensee documented this procedural deficiency in CR M3-97-2886.

The failure to vent the RSS piping is an apparent violation of TS 4.5.2.b.1. (VIO 423/97-206-21)

(c) Inadequate procedure for verification of charging lineup

TS 4.5.2.b.2 requires that Modes 1, 2, 3, and 4, the position of each valve in the ECCS system flow path that is not locked in position be verified every 31 days. The team reviewed procedure SP 3608.4, "High Pressure Safety System Vent and Valve Lineup Verification," Rev. 4. The purpose of this procedure is to perform TS 4.5.2.b.2 requirements for valves in the charging system. The team found that the procedure was deficient because it did not include all charging system valves in the ECCS flow path. The procedure did not verify the position of valves 3CHS*MV8438A,

3CHS*MV8438B, 3CHS*MV8438C, 3CHS*MV8468A, 3CHS*MV8468B, 3CHS*LCV112D, 3CHS*LCV112E, 3CHS*V49, 3CHS*V50, 3CHS*V51, 3CHS*V706, 3CHS*V44 and 3CHS*V707. These valves are not locked in position. The licensee stated that the position of these valves was reviewed during the monthly performance of procedure SP3604.C2, Boration Flow Path Verification, Rev. 4. Although the requirement to verify the position of valves is not specifically addressed in SP 3604.C2, operators would take the appropriate corrective actions if valves were found to be out of their required position. During the inspection, the licensee issued CR M3-97-2832 because the Surveillance Program did not correctly identify the procedure that accomplished TS 4.5.2.b.2 Surveillance Requirements.

The inadequate valve lineup procedure for verifying the charging system valve lineup every 31 days is an example of a violation of Technical Specification 6.8.1, inadequate procedures.
(VIO 423/97-206-13)

8.2 Inservice Pump and Valve Testing

8.2.1 Scope of Review

Review the procedures that test the following pumps and valves to verify that they were tested in accordance with the ASME Boiler and Pressure Vessel Code, Section XI.

- Charging pumps 3CHS*P3A,B,C
- Charging pump cooling pumps 3CCE*P1A,B
- Charging pump discharge check valves 3CHS*V46,47,48
- Charging pump injection header check valve 3SIH*V5
- Charging pump suction valves 3CHS*LCV112B,C,D,E
- Charging pump suction check valve 3CHS*V261
- Charging pump injection valves 3SIH*MV8801A,B
- Charging pump cooling pump discharge check valves 3CCE*V13A,B
- Cold-leg injection check valves 3RCS*V29,70,106,145
- Charging pump miniflow valves 3CHS*MV8511A,B and 3CHS*MV8512A,B

8.2.2 Findings

Valves 3CHS*LCV112D, 3CHS*LCV112E, 3CHS*V261, 3CHS*MV8511A, 3CHS*MV8511B, 3CHS*MV8512A and 3CHS*MV8512B are not classified as ASME Boiler and Pressure Vessel Code, Section XI, IWV-3420, Category A, valves, and, therefore, not tested for seat leakage. These valves are required to be classified as Category A valves and tested for seat leakage because during the recirculation mode of operation, seat leakage past these valves would accumulate in the RWST causing an increase in the radiation dose in the control room and at the site boundary. TS 4.0.5 requires that pumps and valves be tested in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

TS 6.8.4.a requires that a program be established to reduce leakage from systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The program is required to include integrated leak test requirements for the charging part of the chemical and volume control system. To meet TS 6.8.4.a requirements, the valves above are required to be periodically checked for seat leakage.

Further, the NRC had previously issued IN 91-56, "Potential Radioactive Leakage to Tank Vented to Atmosphere," that related directly to this issue. Because the licensee did not utilize FSAR leakage rate assumptions in its analysis of the IN, an opportunity to identify this programmatic deficiency was missed.

Calculation 746P(R), "ECCS Leakage Outside Containment," dated May 12, 1997, stated that during ECCS operation, the estimated leakage rate of fluid from the containment to areas outside of the containment is approximately 1 gallon per hour (gph). FSAR Table 15.6-9, "Assumptions Used For the Radiological Consequences of a LOCA Analysis," states that the maximum post-LOCA equipment leakage rate is assumed to be two times the maximum operation leakage rate for a total leakage rate of approximately 2 gph. The licensee's evaluation of IN 91-56 stated that a leakage rate of 50 gpm (3000 gph) to the RWST from the ECCSs for a 24-hour period was acceptable. This leakage significantly exceeded FSAR leakage rates. At the end of the inspection period, the licensee initiated CR M3-97-3218 to reevaluate valve leakage to the RWST. CR-M3-97-3218 listed the following valves that are not seat leak tested.

3CHS*LCV112B,C,D,E	3SIH*MV8920	3SIH*V261	3SIL*V3
3CHS*MV8511A,B	3SIH*MV8814	3SIL*MV8812A,B	3SIL*V9
3CHS*MV8512A,B	3SIH*MV8813	3SIH*MV8806	
3SIH*V11	3RHS*V43		

The failure to leak check certain valves does not meet the requirements of 10 CFR 50.55a and is an apparent violation of TS 6.8.4. (EEI 423/97-206-22)

8.3 Conclusions

Both the RSS air injection issue and the inadequate program for control of leakage outside containment are significant because of their potential impact on safety. Both issues could have been detected earlier if the licensee had properly evaluated generic information it had available. The inadequacy of the valve lineup procedure used to verify, in accordance with TS, the charging valve lineup every 31 days was indicative weaknesses in the licensee's CMP.

9.0 Entrance and Exit Meetings

Upon arrival onsite, the team conducted an entrance meeting to formally brief the licensee on the scope and duration of the inspection. A partial list of persons who attended the entrance is contained in Appendix B.

After completing the on-site inspection, the team conducted an exit meeting with the licensee on September 24, 1997, that was open to public observation. During the exit meeting the team leader presented the results of the inspection. A partial list of persons who attended the exit meeting is also contained in Appendix B. The team leader also presented the findings and answered questions concerning the inspection at a meeting with the public at the Waterford town hall on the evening of September 24, 1997.

Appendix A

List of Apparent Violations, Unresolved Items, and Inspector Followup Items

This report categorizes the inspection findings as apparent violations (VIO), apparent violations being considered for escalated enforcement (EEI), unresolved items (URIs) or inspector followup items (IFI) in accordance with Chapter 610 of the NRC Inspection Manual. An apparent violation is a matter about which the Commission has concluded there is enough information to conclude a violation of a legally binding requirement has occurred. The violation is classified as apparent until the NRC assigns a severity level and the licensee is given the appropriate chance to respond to the NRC's determinations. A URI is a matter about which the Commission requires more information to determine whether the issue in question is acceptable or constitutes a deviation, nonconformance, or violation. The NRC may issue enforcement action resulting from its review of the identified URIs. An IFI is a matter for which additional information is needed that was not available during the inspection.

Item Number	Finding Type	Section	Title
423/97-206-01	VIO	2.1.2(a) 6.1.2(c)	Violation for failure to properly update FSAR in accordance with 10 CFR 50.71
423/97-206-02	VIO	2.1.2(b) 2.1.2(d) 2.1.2(e)	Violation for failure to follow procedures in accordance with TS 6.8.1
423/97-206-03	VIO	2.1.2(c) 5.1.2	Violation for failure to comply with ASME Code of record as required by 10 CFR 50.55a
423/97-206-04	URI	2.1.2(f)	Unresolved item to resolve check valve single failure requirements
423/97-206-05	URI	2.2.2.(2)	Unresolved item concerning qualification of RSS isolation valve seats
423/97-206-06	VIO	2.3.2	Violation of 10 CFR 50.49 for failure to have certain main steam valves in the environmental qualification program
423/97-206-07	URI	3.3.2(a)	Unresolved item concerning the adequacy of the unit's voltage surge protection
423/97-206-08	IFI	3.3.2(b)	Inspection followup item concerning cable short-circuit qualification
423/97-206-09	URI	3.5.2(a)	Unresolved item concerning 1-hour battery connection
423/97-206-10	IFI	3.5.2(b)	Inspection followup item to verify labeling of hydrogen analyses indicators
423/97-206-11	URI	3.6.2(a)	Inspection followup item on licensee to evaluate 1.8 second delay in undervoltage protection

423/97-206-12	VIO	4.1.2(a) 4.1.2(c) 7.1.2	Inadequate corrective action to correct identified discrepancies in accordance with 10 CFR Part 50, Appendix B, Criterion XVI
423/97-206-13	VIO	4.1.2(b) 4.2.2 6.2.2(a) 6.2.2(b) 8.1.2(c)	Failure to have adequate procedures in accordance with TS 6.8.1
423/97-206-14	IFI	4.3.2(b)	Inspection followup item to correct calculation errors in accordance with CR M3-97-3169
423/97-206-15	URI	5.2.2(a)	Unresolved item concerning pipe support analysis
423/97-206-16	VIO	5.2.2(b)	Violation of 10 CFR Part 50, Appendix B, Criterion III
423/97-206-17	URI	5.3.2	Unresolved item concerning the effects of thermal expansion on rigidly restrained steel
423/97-206-18	IFI	6.1.2(a)	Inspection followup item concerning the adequacy of the licensee's proposed change to the bases of the pressurizer level TS
423/97-206-19	IFI	6.1.2(b)	Inspector followup item concerning the resolution of charging pump area temperature inconsistencies
423/97-206-20	EEI	8.1.2(a)	Apparent violation of 10 CFR Part 50 Criterion XVI for failure to identify and take corrective actions for air in the RSS piping
423/97-206-21	VIO	8.1.1(b)	Violation for failure to vent RSS piping in accordance with TS 4.5.2.b.1
423/97-206-22	EEI	8.2.2	Apparent violation of TS 6.8.4 and 10 CFR 50.55a for failure to leak check certain valves

Appendix B

Entrance & Exit Meeting Attendees

NAME	ORGANIZATION
Neil Carns+	NU Chief Nuclear Officer
Dave Goebel+	NU VP Nuclear Oversight
Dave Amerine+	NU VP Nuclear Engineer and Support
Evan Woollacott*	NEAC
Terry Concannon+	NEAC
Richard Laudenat*	NU-ICAVP
Barry Pinkowitz*	MP3-OPS
Robert Andren*	MP3-Design
Thomas McCarthy	MP3-ICAVP Lead
Denny Hicks*	MP3 Director
Joe Fugere	Mgr ICAVP
Raymond Necci	NU-CAMP
Michael Brothers	MP3
Martin Bowling	MP2
Harry Miller*	MP2
Gil Olsen*	MP3
William Travers+	NRC/Director, SPO
Eugene Imbro+	NRC/Deputy Director, ICAVP, SPO
Peter Koltay	NRC/ICAVP,SPO
Steve Reynolds	NRC/Chief, ICAVP,SPO
Jim Luehman	NRC Team Leader, ICAVP,SPO
Victor Ferrarini*	NRC/Contractor
Don Prevatte*	NRC/Contractor
Stephen Tingen*	NRC/ICAVP,SPO
Omar Mazzoni*	NRC Contractor
Harold Eichenholz	NRC/ICAVP,SPO
Tony Cerne+	NRC Senior Resident Inspector
Beth Korona*	NRC Resident Inspector

*Attended Entrance Meeting only
+Attended Exit Meeting only

Appendix C

List of Documents Reviewed

1. PROCEDURES

a. Abnormal Operating Procedures

-- AOP3555, Rev 7, Reactor Coolant Leak
-- AOP3560, Rev 3, Loss of Service Water
-- AOP3561, Rev 4, Loss of Reactor Plant Cooling Water
-- AOP3566, Rev 4, Immediate Boration

b. Emergency Operating Procedures

-- EOP35 E-1, Rev. 14, Loss of Reactor Secondary Coolant
-- EOP35ES-1-1, Rev. 12, SI Termination
-- EOP35ES1-3, Rev. 6, Transfer to Cold-Leg Recirculation
-- EOP35E-3, Rev. 13, Steam Generator Tube Rupture
-- EOP3503, Rev 11, Shutdown Outside Control Room
-- EOP3506, Rev 5, Loss of All Charging

c. Operating Procedures

-- OP3203, Rev 15, Plant Startup
-- OP3252 Rev 3, Change 1, Operator Aids
-- OP3301D, Rev 11, Change 3, Reactor Coolant Pump Operation
-- OP3304A Rev 25, Charging and Letdown
-- OP3314J, Rev 4, Auxiliary Building Emergency Ventilation and Exhaust
-- OP3326, Rev 18, Change 11, Service Water System
-- OP3330A Rev 13, Change 1, Reactor Plant Component Cooling Water
-- OP3330D Rev 5, Change 1, Charging Pump Cooling
-- OP33531s, Rev. 0, Change 2, Instrument and Service Air Panel Annunciator Response
-- OP3353.MB1A-MB4C, Main Board Annunciator Response
-- OP3353VP1A-C, Main Ventilation and Air Conditioning Annunciator Response

d. Special Procedures

-- SP3601F.4, Rev. 9, RCS Pressure Isolation Valve Test
-- SP3604A.1, Rev 9, 7/28/97, Charging Pump A Operational Readiness Test
-- SP3604A.3, Rev. 8, Charging Pump C Operational Readiness Test
-- SP3604A.5, Rev 10, 8/7/97, Chemical and Volume Control System Valve Operability Test
-- SP3608.4, Rev. 2, High Pressure Safety Injection Valve Lineup Verification
-- SP3608.6, Rev. 11, Safety Injection System Valve Operability Test.
-- SP3612B.4, Rev 12, Change 3, 4/17/97, Containment Local Leak Rate Test Type C Penetrations
-- SP3626.13, Rev 15, Service Water heat Exchangers Fouling Determination
-- SP3646A.17, Rev. 9, Train A ESF With LOP Test
-- SP3859,Rev 3, 8/29/96, RWST Boron Concentration

e. Nuclear Group Procedures

NGP 6.12, Rev. 1, Evaluation of Replacement Items

f. Maintenance Procedures

-- MP3783 EA Rev 3, Component Cooling Pumps Motor Replacement for Fire Protection

g. Miscellaneous Procedures

-- EN 311121, Rev 6, 12/10/96, IST Pump Operational Readiness Evaluation
-- CP 807/2807/3807AA, Rev 0, 9/13/93, Boron Analysis

2. DRAWINGS

-- 12179-EM-102A-F, Reactor Coolant System
-- 12179-EM-103A, Rev 15, 5/30/97, Reactor Coolant Pump Seals
-- 12179-EM-104A-D, Chemical & Volume Control
-- 12179-EM-105A, Rev 12, 6/27/97, Charging Pump Sealing and Lubrication
-- 12179-EM-112A, Rev 25, 2/7/97, Low Pressure Safety Injection
-- 12179-EM-112C, Rev 16, 2/7/97, Low Pressure Safety Injection/Containment Recirculation
-- 12179-EM-113A&B, High Pressure Safety Injection
-- 12179-EM-115A Rev 19 - Quench Spray & H_2 Recombiner
-- 12179-EM-121B Rev 14 - Reactor Plant Component Cooling Water
-- 12179-EM-133B, Rev 34, 4/15/97, Service Water
-- 12179-EM-148A&B, Reactor Plant Ventilation

-- 12179-ESK-5CS Rev 11 - Charging Pump P3A
-- 12179-ESK-5CT Rev 10 - Charging Pump P3B
-- 12179-ESK-5CU Rev 11 - Charging Pump P3C (Train A Power)
-- 12179-ESK-5CV Rev 9 - Charging Pump P3C (Train B Power)
-- 12179-ESK-5DE Rev 16 - RHR Pump P1A
-- 12179-ESK-5DF Rev 16 - RHR Pump P1B
-- 12179-ESK-6PK Rev 13 - VCT Outlet 112B
-- 12179-ESK-6PL Rev 13 - VCT Outlet 112C
-- 12179-ESK-6PM Rev 14 - VCT Outlet 112D
-- 12179-ESK-6PN Rev 11 - VCT Outlet 112E

-- 12179-LSK-9-1C Rev 11 - Reactor Plant Component Cooling Water
-- 12179-LSK-26-2.2C, 2.2D- VCT
-- 12179-LSK-26-2.3A through 2-2.3L - Charging Pumps
-- 12179-LSK-26-2.A Rev 9 Seal Water Isolation
-- 12179-LSK-27-17A - 27-17C - Safety Injection Actuation
-- 12179-LSK-27-7A Rev 13

3. CALCULATIONS

-- P(R) 746, Rev 0, 3/10/82, ECCS System Leakage Outside Containment
-- P(R) 0982, Rev 0, ECCS Pressure Drop Calculations Based on Westinghouse Piping Resistance Criteria
-- P(R)-0983, Rev 0, 4/23/84, NPSH Evaluation for ECCS Pumps RHS, SIH, CHS - Maximum Safeguards
-- P(R)-1057, Rev 0, 12/4/84, Process Requirements for Procurement of Suction Pipe Relief Valves for CHS*P3A, B, C

--	P(R)-1183, Rev 0, 10/3/96, Operating Pressures and Temperatures for the CHS System to be Used As Input in the Stress Data Package
--	P(R) 1183, Rev 0, 10/28/85, Operating Pressures and Temperatures for the CHS System to be Used in the Stress Data Package

--	UR(B)-382-0, Rev 0, 4/6/84, To Determine If Various Valves in the ESF and Auxiliary Building Are Less Than 1.0 X 10^6 Rads
--	UR(B)-393-0, Rev 0, 4/30/84, Lifetime Radiation dose to 3RSS*MOV20A, B, C, and D
--	UR(B)-398-0, Rev 0, 4/30/84, Post LOCA Beta Dose for Equipment in Contact with Containment Sump Water
--	UR(B)-400, Rev 0, Gamma and Beta Equipment Qualification Doses for Normal Operations, Depressurized LOCA and Pressurized LOCA

--	230, Rev 1, 4/18/80, Available NPSH for Varying Recirc. Pump Flow Rate - Reference Elevation
--	232, Rev 2, 4/15/85, Floor Sumps Water Supply as a Function of Floor Water Depth
--	249, Rev 3, 5/15/92, Determination of Maximum Water Level Inside containment Following a LOCA
--	294, Rev 4, 9/9/85, NPSH Available for ECCS Pumps
--	313, Rev 2, 8/19/93, Temperature Response of a Motor to the Peak MSLB Transient Within the MSVB
--	900P(B), Charging Pump and Component Cooling Pumps Areas Ventilation System
--	1130P(b), Temporary Ventilation for CCP Pumps Area During Loss of Primary Ventilation due to Fire

--	86-317-742 GM, Rev 1, 11/15/95, Required Service Water flow to the Charging Pump Lube Oil Coolers - 3CCE*E1A&B
--	90-069-1130 M3, Rev 0, Change 2, 5/22/97, Service Water System - Summary of Westinghouse Heat Exchanger Calculations
--	93-LOE-342E3, Rev 1, 8/23/93, RCP 3RCS*P1B Seal Leakoff Range Flow Loop Uncertainty & High Alarm Setpoint
-	94-ENG-01037-M3, Rev 0, 8/8/94, Charging Pump Coolers 3CCE*E1A&B; Required Service Water Flow With a 78°F SW Temperature

--	3-ENG-181, Rev 0, 11/14/90, Determination of Available NPSH to Charging Pumps During Gravity Boration
--	W3-517-409-RE, Rev 0, 1/4/84, Alternate Means of Cooling the Safety Injection and Charging Pumps
--	SP-3CHS-12, Rev 0, 4/30/85, 3CHS*RV8351 Containment Penetration Z62 Relief Valve
--	FSE/SS-NEV-1924, Rev 0, 5/21/93, SWS PEGISYS Model Uncertainty
--	SE/FSE-C-NEU-0154, Rev 0, Change 1, 6/3/97, Millstone Unit 3 ECCS Evaluation - Future Tech Spec Change Basis
--	SP-3CHS-9, Rev 0, 4/17/85, Charging Pump Suction Line Overpressure Protection Relief Valves - 3CHS*RV8501A, B & C
--	NSP-193-CHS, Rev 0, 12/30/86, CHS System Relief Valve Setpoints
--	SP-3SIH-6, Rev 0, 7/13/85, 3SIH*RV8870 Containment Penetration Z99 Overpressure Relief Valve
--	SP-3CHS-2, Rev 0, 1/27/84, Pressure Set Point 3CHS-PS393A, B, C Charging Pump Aux. L.O. Pump Control
--	3CCE-TIC 37 A/B, Charging Pumps Coolers Outlet Water Temperature Controller, dated 5/20/82
--	12176-NP(F)-677-XD, Rev. 2 inc. CCN 1 thru 3, Small bore pipe stress analysis: CVCS Containment

-- 12176-NP(F)-743-XD, Rev. 2 inc. CCN 1 thru 2, Small bore pipe stress analysis: CVCS Containment
-- 12179-US(B)-245, Rev 0, Change 3, 4/18/97, Branch Flow Rate Analysis for Safety Injection and Containment Recirculation System
-- 12179-US(B)-311, Rev 0, Change 1, 4/18/97, RSS Branch Flow Analysis with Degraded Pump Curve
-- 12179-US(B)-342, Rev 1, Change 2, Recirculation Spray Heat Exchanger UA's

-- 12179-NP(F)-730, Rev. 4, Small Bore Pipe Stress Analysis Charging Pump Cooling, Aux. Bldg. PLI CP-410113 thru 118, 515 thru 519
-- 12179-NP(F)-729 Rev. 3, Small Bore Pipe Stress Analysis: Charging Pumps Cooling Piping Aux. Bldg Iso's CP-410100, 111, 112, 120, 511, 514
-- 12179-NP(F)-2017,-2029,-2030, Rev. 3, Comparison of Calculated Equipment Nozzle Loads with Allowable Values for Charging Pumps 3CHS*P3A, B, and C
-- 12179-NM(S)-677, Rev. 1, Charging Safety Injection Pumps 3CHS*P3A, B, C Embedment Loads
-- 12179-NS(B)-120, Rev. 0, Class 2, Unsleeved Penetration Calculation (part for the high pressure safety injection inside and outside containment)
-- 12179-NM(B)-127-JAK, Rev. 2, inc. CCN 1, Residual Heat Removal Line Pipe Rupture Restraints and Space Frame
-- 12179-NP(B)-X10700, Rev. 1, inc. CCN 1 thru 6, Containment Structure Annulus Piping - ASME Class 1 & 2
-- 12179-NP(B)-X11004, Rev. 1, inc. CCN 1 thru 4, High Pressure Safety Injection (SIH) piping AUX. BIDG. Piping (ASME III Code Cl. 2 piping)
-- 12179-NP(F)Z-110-108, Rev. 2, inc. CCN 1, Design of Pipe Support: 3-SIH-2-PSA108
-- 12179-NP(F)Z-110-013, Rev. 5, inc CCN 1 thru 5, Design of Pipe Support: 3-SIH-2-PSR013
-- 12179-NP(F)Z-110-014, Rev. 3, inc CCN 1, Design of Pipe Support: 3-SIH-2-PSR014
-- 12179-NP(F)Z-110-015, Rev. 3, inc. CCN 1, Design of Pipe Support: 3-SIH-2-PSR015
-- 12179-NP(F)Z-110-017, Rev. 3, inc. CCN 1 thru 2, Design of Pipe Support: 3-SIH-2-PSR017
-- 12179-NP(F)Z-110-018, Rev. 2, inc. CCN 1, Design of Pipe Support: 3-SIH-2-PSR018
-- 12179-NP(F)Z-110-623, Rev. 2, inc. CCN 1, Design of Pipe Support: 3-SIH-2-PSR623
-- 12179-NP(F)Z-110-624, Rev. 1, inc. CCN 1, Design of Pipe Support: 3-SIH-2-PSR624
-- 12179-NP-SR-007, Rev. 3, P.E. Certified Stress Report for High Head Safety Injection
-- 12179-NP(F)-2707-XD, Rev. 0 inc. CCN 1 thru 2, Comparison of Calculated Equipment Nozzle Loads with their Allowable Values for: Charging Pump Cooling Pump 3CHS*P1A
-- 12179-NM(B)-534-IE, Rev. 1, Refueling Water Storage Tank Design & Analysis
-- 12179-SEO-C30.15, Rev. 1, Refueling Water Storage Tank Foundation Evaluation for Outlet Loads-3QSS*TK1
-- 12179-NM(S)-514-IE, Rev. 0, Refueling Water Storage Tank Seismic Analysis
-- 12179-NP(B)-X10717, Rev. 1, inc. CCN 1 thru 7, Pipe Stress Analysis: Containment Structure Annulas Piping ASME Class 1& 2
-- 12179-NP(B)-X10730, Rev. 1, inc. CCN 1 thru 4, Containment Structure Annulus Piping SIH to Cold-Leg Loop 3 ASME Class 2 & 3 Piping
-- 12179-NP(B)-X10702, Rev. 1, inc. CCN 1 thru 9, Containment Structure Annulus Piping, High Pressure Safety Injection to Cold-Leg Loop 2 & 4
-- 12179-C24.1, -C24.23, 2/14/78, Buried Piping-Conc. Encasement EC-24A,B and Service Water Encasement at Col. Line 44.3 ESF building
-- 12179-P(R)-1179, Rev. 1, Documentation of Operating Temperature and Process used as input to SDP-SIH
-- 03705-S52.31, Rev. 0, Steel Load Re-verification Program Annulus Pipe Rack Framing Analysis Containment Building

4. PLANT DESIGN CHANGE REQUESTS

-- DCR M3-97008, Rev 0, 3/18/97, Replacement of ASCO SOVs
-- DCR M3-96077, Rev 0, 3/24/97, ECCS Orifices and Throttle Valves
-- DCR M3-97527, Rev 0, 2/28/97, Replacement of ECCS Flow Measurement Orifices 3SIH*FE917, 918, and 922 with Similar Calibrated Orifices
-- PDCR MP3-93-163, Rev 0, 9/13/93, Modification to Swing Check Valve 3CHS-V165
-- PDCR MP3-95-020, Rev 0, 4/17/95, 3CHS*MV8507 A/B Disk Modification
-- PDCR MP3-95-003, Rev 0, 3/2/95, RCP No 1 Seal Injection Valves Replacement
-- RIE # PSE-MP3E-94-101, 2/1/95, Replacement of Relief Valve 3CHS*RV8119
-- PDCR MP3-93-126, Rev 0, 8/7/93, CHS & RCS Valve Yoke Bolts Replacement
-- PDCR M3-91-067, Rev 0, 4/4/91, Change Auto Makeup Setpoint from 120 gpm to 80 gpm
-- MMOD M3-96606, Rev 0, 7/17/97, Revision of Lube Oil on Charging Pump Sealing and Lubrication P&ID
-- PDCR MP3-89-095, 6/4/89, "C" Charging Pump Orifice Installation

5. ADVERSE CONDITION REPORTS

ACR 3577 (unit 2)

6. SPECIFICATIONS

-- Engineering Specification SP-M3-EE-0333, Rev 0, Environmental Conditions for Equipment Qualification
-- Specification 2362.200-164, 1/10/85, Mechanical Equipment Environmental Qualification

7. LICENSING DOCUMENTS

-- FSAR Chapters 6, 9, and 15
-- FSAR Section 1.2.10, Engineered Safety Features
-- FSAR Table 1.3-1, Design Comparison
-- FSAR Section 1.8 Conformance to NRC Regulatory Guides
-- FSAR Section 1.8N, NSSS conformance to NRC Regulatory Guides
-- FSAR Section 3.1, Conformance with NRC General Design Criteria
-- FSAR Section 3.2.4, Other Classification Systems
-- FSAR Appendix 3B, Environmental Design Conditions
-- Millstone 3 Technical Specifications
-- FSAR Section 5.4.1.2.2, [RCP] Pump Assembly Description
-- FSAR Section 5.4.1.3.10, Shaft Seal Leakage
-- FSAR Section 5.4.1.3.11, Seal Discharge Piping
-- FSAR Section 5.4.3, Reactor Coolant Piping
-- FSAR Question Q480.3 and Answers
-- Millstone 3 Technical Requirements Manual, Change 39

8. LICENSEE EVENT REPORTS (LERs)

-- LER 89--12, 7/5/89, Containment Leakage in Excess of Limits Due to Valve Leakage
-- LER 89-022, 10/25/89, Valve Stroke Time Testing in the Wrong Direction Due to Transcription Error
-- LER 91-011, 5/10/91, Both Trains of high Pressure Safety Injection System inoperable Due to Relief Valve Leakage
-- LER 92-026, 12/3/92, Both Trains of High Pressure Safety Injection Inoperable
-- LER 93-006, 6/21/93, Inadequate Surveillance Testing of High Pressure Safety injection Check Valves

- -- LER 94-007, 5/13/94, Violation of Engineered Safety Feature Response Time of Quench Spray System
- -- LER 94-010, 8/24/94, Both Trains of Charging Inoperable Due to Procedural Deficiency
- -- LER 96-028, & 01, 12/13/96, Potential Overcooling of Charging Pump Lube Oil System Due to Failure of Air-Operated Temperature Control Valves
- -- LER 96-029, 9/29/96, Functional Deficiency in the Setting of the Emergency Core Cooling System Throttle Valve Positions
- -- LER 96-031, 10/6/96, Potential Failure of Safety Related Control Valves Due to Failure of Non-Qualified Air Regulators
- -- LER 96-032, 10/9/96, High Pressure Safety injection Relief Valve Piping Hydrostatic Test Non-Compliance with ASME Code Due to Personnel Error
- -- LER 96-044, 12/4/96, Qualification of Containment Systems Following a Design Basis Accident
- -- LER 97-008, 2/22/97, Failure to Enter Technical Specification 3.0.3 Action Statement for MSIV Closure
- -- LER 97-018, 3/7/97, Technical Specification Parameter Compliance

9. OTHER DOCUMENTS/MANUALS REVIEWED

- -- Henry Pratt Company Drawing of Valve 3RSS-MOV23A, B, C, and D, S&W Dwg. No. 25212-29164, Sh 110, Rev 1
- -- Lonergan "LCT" Series Relief Valves Vendor Manual, Document ID # OIM-186-1A, /Rev 1G 8/2/96
- -- Charging/Safety Injection Pumps Vendor Manual, Document ID # 25212-001-019A, Rev L, 5/20/97
- -- Electrical Equipment Qualification Program Manual, Rev 1
- -- Memo from: D. T. McDaniel to M. P. Pearson: Operability of CHS Pumps Without Fans 3HVR*12/14 in Service in Mode 5, dated October 10, 1992\
- -- Letter from: W. J. Johnson, Westinghouse to R. C. Jones, NRC: Use of 2700°F PCT Acceptance Limit in Non-LOCA Accidents
- -- PORC/SORC Meeting Minutes - March - August 1997
- -- Fire Protection Evaluation Report
- -- Technical Bulletin 79-6, Pacific Centrifugal Charging/Safety Injection Pumps, 9/25/79
- -- Installation, Operation and Maintenance Manual No. 25212-001-019A, Charging/Safety Injection Pumps, Rev. L
- -- Installation, Operation and Maintenance Manual, OIM-042, 002A, Charging Pump Cooling Pump, Rev 2F
- -- Work Order M3-94-03094 dated 7/27/95, Replace Antirotation Key on 3CHS*MV8111A
- -- Work Order M3-93-19263 dated 8/30/93, Replace Antirotation Key on 3CHS*MV811D

Appendix D

List of Acronyms

ACR(s)	adverse condition report(s)
AOP(s)	abnormal operating procedure(s)
AR	action request
ASME	American Society of Mechanical Engineers
ASP	auxiliary shutdown panel
BOM	bill of materials
BTP	Branch Technical Position
CCE	charging pump cooling
CCN	calculation change notice
CFR	*Code of Federal Regulations*
CR(s)	condition report(s)
DBE	design basis event
DDR	design deficiency report
ECCS	emergency core cooling system
EDG	emergency diesel generator
EEI	escalated enforcement item
EEQ	electrical equipment qualification
EOP(s)	emergency operation procedure(s)
EMI/RFT/	electromagnetic/radio frequency interference/electrostatic discharge
EP	ethylene propylene
EPRI	Electrical Power Research Institute
EPT	ethylene propylene terpolymer
EQ	equipment qualification
ESF	engineered safety feature
FSAR	Final Safety Analysis Report
FSARCR	Final Safety Analysis Report Change Request
GDC	general design criterion/criteria
gph	gallons per hour
gpm	gallons per minute
HP	high-pressure
ICAVP	Independent Corrective Action Verification Program
IEEE	Institute of Electrical and Electronics Engineers
IN(s)	Information Notice(s)
IPE	individual plant examination
IST	inservice testing
LCO	limiting condition for operation
LER(s)	licensee event report(s)
LLRT	local leak rate testing
LOCA	loss-of-coolant accident
LOOP	loss of offsite power

LOP	loss of power
MAE	maximum abnormal excursion
MCC	motor control center
MCR	main control room
MNE	maximum normal excursion
MOV	motor-operated valves
MSLB	main steam line break
MSVB	main steam valve break
NGP(s)	Nuclear Group Procedure(s)
NNECO	Northeast Nuclear Energy Company
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OP(s)	operating procedure(s)
P&ID	piping & instrumentation diagrams
PMMS	preventive maintenance management system
PORV(s)	power operated relief valve(s)
QA	quality assurance
RCP(s)	reactor coolant pump(s)
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RIE	Replacement Item Evaluation
RPCCW	Reactor plant component cooling water
RPS	reactor protection system
RSS	recirculation spray system
RSST	reserve station service transformer
RWST	refueling water storage tank
SI	safety injection
SSFI	safety system functional inspection
SW	service water
TR(s)	trouble report(s)
TRM	Technical Requirements Manual
TS(s)	technical specification(s)
URI(s)	unresolved item(s)
VCT	volume control tank
Vdc	volts, direct-current